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NUCLEAR SCIENCE ABSTRACTS

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The printing of this publication has been approved by the Director of the Bureau of the Budget, July 30, 1957.

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SUPPLEMENT

CHEMISTRY

701 ANL-4427(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-II SUMMARY REPORT FOR OCTOBER, NOVEMBER, AND DECEMBER 1949. J. R. Gilbreath and O. C. Simpson, comps. July 28, 1950. Decl. with deletions Feb. 11, 1957. 112p. Contract W-31-109-eng-38. $18.30(ph OTS); $6.00(mf OTS).

702 CF-52-8-212(Del.)
Oak Ridge National Lab., Tenn.
PHYSICAL PROPERTY CHARTS FOR SOME REACTOR FUELS, COOLANTS, AND MISCELLANEOUS MATERIAL. Aug. 28, 1952. Decl. with deletions Feb. 20, 1957. $1.80 (ph OTS); $1.80(mf OTS).

Charts of physical properties of some reactor fuels, coolants, and miscellaneous materials have been prepared. Densities, volume expansion coefficients, heat capacities, thermal conductivities, viscosities, and melting temperatures are tabulated. (L.T.W.)

703 CF-54-6-188(Del.)
Oak Ridge National Lab., Tenn.
PHYSICAL PROPERTY CHARTS FOR SOME REACTOR FUELS, COOLANTS, AND MISCELLANEOUS MATERIALS. (Fourth Edition). June 21, 1954. Decl. with deletions Feb. 20, 1957. 15p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

Charts of physical properties of some reactor fuels, coolants, and miscellaneous materials have been prepared. Densities, volume expansion coefficients, heat capacities, thermal conductivities, viscosities, electrical resistivities, surface tensions, and melting temperatures are tabulated. (L.T.W.)

704 CF-55-11-123
Oak Ridge National Lab., Tenn.
THE CAUSTIC DISSOLUTION OF UNIRRADIATED URANIUM–ALUMINUM ALLOY FUEL IN A TRICKLE-TYPE DISSOLVER. D. L. Foster. Nov. 25, 1955. Decl. Apr. 4, 1957. 14p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

The dissolution of U – Al alloy fuels in sodium hydroxide and sodium hydroxide-sodium nitrate dissolvents was tested in a trickle-type dissolver. Dissolution rates as high as 233 kg of Al/day were obtained for 5.37% U extruded slugs packed in a 5.5-in.-id. column to a height of 6 ft. The dissolvent in this case was 6.0M sodium hydroxide and the product was 2.5 to 3.0M in sodium aluminate. With a 3.0M sodium nitrate and 6.0M sodium hydroxide dissolving solution, the reaction rate for the same height of packing was increased over that for 6.0M caustic by a factor of approximately two. The sodium nitrate was added for the purpose of suppressing the evolution of hydrogen, but this effect was not accomplished. The resulting gases still contained approximately 70% H by volume. (auth)

705 HW-44584
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
THE SUPPRESSION OF CHLORIDE VOLATILIZATION IN NITRIC ACID DISTILLATION. R. L. Moore. July 27, 1956. Decl. Mar. 14, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Chloride-induced corrosion in Purex and similar processes may be reduced by the addition of $Hf(NO_3)_2$. (D.E.B.)

706 ISC-421(Del.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN CHEMISTRY FOR JULY, AUGUST, AND SEPTEMBER 1953. Dec. 1, 1953. Decl. with deletions Feb. 26, 1957. 23p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

The following investigations are reported: melting points of Sm and Yb; reaction of bromate with bromide in fused nitrates, catalyzed by Zn^{+2}; formation of complexes between perchlorate and Tl^{+3}; the Tl – Fe reaction; decomposition of $GaCl_2$; ferromagnetic transition of CrF_3 at 77°K; titrimetric determination of Zn in U – Zn alloys; determination of sulfate; spectrochemical determination of Gd, Sm, and Eu in Zr; determination of fission products in U – Cr eutectic; determination of Ta in Nb; separation of nuclear fuel and by-products with liquid Ag or La; photonuclear processes in Os; Szilard-Chalmers type reactions in Os; exchange reactions between Mn^{++} and MnO_4^{-2} and Cl^- and $PtCl_4^{-2}$; structure and residual entropy of ice; and hydrogen bonding in H_2O and D_2O. (J.E.D.)

707 KAPL-1534
Knolls Atomic Power Lab., Schenectady, N. Y.
SOME QUALITATIVE OBSERVATIONS OF ZIRCONIUM-WATER REACTIONS. D. C. Layman and H. L. Mars. Apr. 23, 1956. Decl. Apr. 11, 1957. 23p. Contract W-31-109-Eng-52. $0.35(OTS).

Self-sustaining and self-propagating chemical reactions between Zircaloy and water have been demonstrated. High-speed motion pictures illustrating the phenomena involved have been made. Qualitative results of a test program limited to experiments at atmospheric pressure, stagnation and at temperatures in the subcooled range from 40 to 150°F are presented. Samples were heated by electrical power inputs of increasing exponential periods in the range of 2 to 20 msec. The chemical reaction was initiated very close to the melting point of Zircaloy, proceeded with varying rates, and sustained itself to complete loss of sample in some cases. The reaction showed the characteristics of a combustion process rather than an explosion. The hypothesis is advanced that the rate of the chemical reaction is dictated by the rate of diffusion of the reactants into a reacting zone, and that temperature sensitivity is not the rate-controlling parameter. (auth)

708 KLI-447
Carbide and Carbon Chemicals Corp. [K-25 Plant], Oak Ridge, Tenn.

REPORT ON THE HANDLING AND HYDROLYSIS OF CHLORINE TRIFLUORIDE, WITH A DESCRIPTION OF APPARATUS AND PROCEDURE. Roger L. Jarry and Wallace Davis, Jr. Jan. 13, 1950. Decl. Mar. 29, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A summary of observations made in handling small amounts of ClF_3 and hydrolyzing these samples for quantitative chemical analysis is given. (L.T.W.)

709 M-356(Book I, II, III)

American Cyanamid Co. Stamford Labs., Stamford, Conn.

OPERATING INSTRUCTIONS—PROCESS FOR THE PRODUCTION OF [CRYSTALLINE BORON] PRODUCT 891A. R. D. Anderson, W. M. Bowes, E. L. Carpenter, G. I. Cathers, J. K. Dixon, K. M. Milton, B. C. Rowley, R. C. Ryder, M. Sonnino, and R. T. Swain. July 1946. Decl. Mar. 6, 1957. 353p. Contract [W-7401-eng-91]. $69.90 (ph OTS); $21.60(mf OTS).

These three reports are bound separately but are cataloged as a unit.

A process is described for the production of approximately 1 kg/day of crystalline boron from a boron trifluoride–dimethyl ether complex. An over-all picture of the process is given by the master flowsheet. The equipment and procedure used in carrying out these operations are discussed in detail. Schematic sketches, wiring diagrams, and photographs of the equipment are included. (C.H.)

710 M-2433

Standard Oil Co. of Indiana, Chicago.

FORMATION OF METHYL BORATE FROM THE METHYL BORATE–BORON FLUORIDE COMPLEX. D. A. McCaulay and W. L. Rittschof. Sept. 4, 1945. Decl. Mar. 7, 1957. 6p. Contract W-7418-eng-41. $1.80(ph OTS); $1.80(mf OTS).

The possible displacement of the equilibrium of the reaction $B(OMe)_3 \cdot 2BF_3 + 2Me_2O \rightleftharpoons B(OMe)_3 + 2BF_3 \cdot Me_2O$ by the presence of free methyl ether was investigated. Methyl ether was bubbled through boiling methyl borate–boron fluoride complex, and the vapors were rectified. It was found that the overhead product consisted mainly of methyl borate and that the residue consisted of methyl ether–boron fluoride complex. It is concluded that the reaction goes to completion under these conditions and that the occurrence of this reaction could account for the high losses of boron from the fractionation unit which have been observed. (L.T.W.)

711 MCW-110

Mallinckrodt Chemical Works, St. Louis.

BEAKER EXPERIMENT TO DETERMINE THE OPTIMUM CONCENTRATIONS OF Na_2CO_3 AND $NaHCO_3$ IN THE ADJUSTED LIQUORS FOR THE C-3 PROCESS [CARBONATE LEACHING PROCESS]. W. G. Weber. Aug 21, 1948. Decl. Mar. 28, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

712 MITG-A50

Massachusetts Inst. of Tech., Watertown, Mass. Mineral Engineering Lab.

PRODUCTION OF SULFURIC ACID AND FERRIC SULFATE BY AIR OXIDATION OF SO_2 AND FERROUS SULFATE IN TOWERS. John J. Brunner, Charles O. Jackson, and Robert R. Porter. Sept. 27, 1948. Decl. Mar. 9, 1957. 34p. Contract W-7405-Eng-85. $6.30(ph OTS); $3.00(mf OTS).

Sulfur dioxide and air, when passed into a dilute $FeSO_4$ solution, will make the $Fe(SO_4)_3$ and H_2SO_4 needed for leaching the Rand ores. This investigation concerns the use of packed towers for these reactions to improve economy and practicality of the process. The reactions can be completed in a packed tower only at low flow rates, but if the SO_2 is absorbed in a packed tower by the dilute ferrous sulfate solution and the effluent run into an air-agitated pulp, the air normally required to keep the ore in suspension will furnish the O needed to produce the leaching reagents. (auth)

713 NAA-SR-1433

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

GENERAL CHEMISTRY QUARTERLY PROGRESS REPORTS FOR PERIODS OCTOBER–DECEMBER 1954 AND JANUARY–MARCH 1955. Eugene L. Colichman and E. E. Motta, eds. Mar. 15, 1956. Decl. Apr. 5, 1957. 265p. Contract AT-11-1-GEN-8. $1.60(OTS).

Radiation decomposition data were obtained on terphenyl systems irradiated with 1-Mev electrons at 400°C. MTR in-pile irradiations of terphenyls were continued. The compressed KBr-pellet infrared method was developed for analysis of damaged o- and m-terphenyl. The pyrolytic products of p-terphenyl were identified. A procedure is outlined for the spectrographic analysis of m-polyphenyls. Analytical procedures are also presented for the determination of O_2 in He, Zr, and NaK, U in Th and Zr, Bi, Pb, and Fe in Pb–Bi alloys, and Fe in chromium steel. (For preceding period see NAA-SR-1205.) (C.W.H.)

714 NYO-218

Girdler Co. Gas Processes Div. Process Development and Research Labs., Louisville, Ky.

SUPPLEMENTARY LABORATORY STUDIES. (Laboratory Studies Relating to the Separation of Boron 10). Polymer Project. July 3, 1953. Includes Reports: (T1.17-1) by James T. Bradbury. July 1, 1953; (T8.10.-1-3) by Kenton Atwood. July 2, 1953; and (T9.40.-2-3) by W. M. Keely and Kenton Atwood. July 2, 1953. Decl. Mar. 12, 1957. 21p. Contract AT(30-1)-1506. $4.80(ph OTS); $2.70 (mf OTS).

Studies in connection with the distillation of $(Ch_3)_2O–BF_3$ complex, the reaction of $(C_2H_5)_2O$ with the complex, and removal of HCl from $BCl_3–H_2$ mixtures are described. (T.R.H.)

715 NYO-1269

Hooker Electrochemical Co., Niagara Falls, N. Y.

THE HYDROGEN REDUCTION OF BORON TRICHLORIDE ON ALLOY WIRES. E. A. Belmore and A. M. Teller. Aug. 8, 1956. Decl. Apr. 12, 1957. 30p. Contract AT(30-1)-1524. $1.00(OTS).

The process for the manufacture of isotopically enriched pure boron was reactivated late in 1953 with the Hooker Electrochemical Company as operating contractor. In order to become familiar with the chemical and mechanical aspects of the process, and to try to develop improvements, a laboratory study of each processing step was started and, in some cases, continued during the operation of the plant. This report covers the work done on the step for reducing boron trichloride with hydrogen on a hot wire inside a glass tube. From the limited amount of work done, no probability of making any major reduction in the cost of this step was discovered. A number of minor improvements were developed which made operation less difficult and which contributed to a longer operating life and hence a greater productivity and lower unit of cost of the expensive alloy wires. It is believed, however, that increased capacity for this product should be secured by developing a metal reactor containing a multiple number of wires or rods rather than by increasing the number of these standard glass units containing a single wire. (auth)

716 NYO-1317

Mallinckrodt Chemical Works, St. Louis.

MCW [MALLINCKRODT SF] ACCOUNTABILITY PROGRAM. Progress Report. C. H. Walden. Jan. 2, 1951. Decl. Feb. 28, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

Accounting progress, analytical and theoretical development, and production aspects of the SF materials accounting program at Mallinckrodt Chemical Works are briefly reviewed. (L.T.W.)

717 TID-10163
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1954. C. J. Rodden. Feb. 1955. Decl. Aug. 8, 1957. 90p. $0.60(OTS).

Chemistry of Feed Materials. Studies in boron production by electric arc reduction of BCl_3, and in carbonate leaching of U from MgF_2 slag are reported. Analytical Chemistry. The derivative polarography of U in concentrated solutions was investigated. An improved arrangement for the shotgun analysis of uranium is described. Preliminary experiments on a vacuum fusion analysis of U using the iron bath technique are reported. (T.R.H.)

718 WAPD-FE-788
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
A COMPARATIVE SUMMARY OF CHEMICAL AND PHYSICAL ANALYSES OF THE INITIAL SHIPMENTS OF MALLINCKRODT UO_2 POWDERS FOR PHYSICS CRITICAL EXPERIMENTS. J. Glatter. May 6, 1955. Decl. Mar. 19, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Analytical Procedures

719 CN-3678
Argonne National Lab., Lemont, Ill.
THE LANTHANUM FLUORIDE-POTASSIUM BROMATE METHOD OF ANALYSIS FOR PLUTONIUM(III, IV). J. R. Gilbreath. Nov. 6, 1946. Decl. Apr. 4, 1957. 6p. Contract W-31-109-eng-38. $1.80(ph OTS); $1.80(mf OTS).

A suitable procedure for the analysis of $BiPO_4$ Process solutions for relative amounts of Pu in different oxidation states was developed. The method consists of precipitation of LaF_3 in presence of 0.02M $KBrO_3$ holding oxidant. (C.W.H.)

720 HW-25860
[Hanford Works, Richland, Wash.]
SPECTROCHEMICAL DETERMINATION OF IRON AND SILICON IN URANIUM. J. L. Daniel. Oct. 7, 1952. Decl. Apr. 3, 1957. 7p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

A spectrochemical method for the determination of trace amounts (~100 ppm) of Fe and Si in U is described. The time required for a complete analysis is 5 hr. (C.W.H.)

721 HW-36788
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
A GAMMA ABSORPTOMETER FOR THE IN-LINE DETERMINATION OF PLUTONIUM OR URANIUM. D. G. Miller and R. E. Connally. June 1, 1955. Decl. Apr. 10, 1957. 45p. Contract W-31-109-Eng-52. $0.30(OTS).

A gamma absorptometer applicable to the continuous determination of Pu or U in process streams has been developed. This instrument utilizes Tm^{170} as the source of gamma photons, a stainless steel container fitted with fluorothene windows as the sample cell, a lead-lined ionization chamber as the detector, and a modified Beckman micro-microammeter as the current measuring system. (L.T.W.)

722 KAPL-572(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
ANALYTICAL CONTROL METHODS FOR PUREX. B. F. Rider—Ruth M. Nolin, ed. Apr. 27, 1951. Decl. with deletions Feb. 25, 1957. 106p. Contract W-31-109-Eng-52. $16.80(ph OTS); $5.70(mf OTS).

The methods used at KAPL for the analytical control of the Purex Process are presented as step-by-step procedures. (C.H.)

723 KAPL-M-ELS-4
Knolls Atomic Power Lab., Schenectady, N. Y.
THE DETERMINATION OF NITROGEN IN ZIRCONIUM. E. L. Shirley and G. R. Smith. June 6, 1956. 5p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Nitrogen in Zr has been determined by a dissolution and micro Kjeldahl technique. Recoveries are shown for known amounts of nitrogen gas after reaction with Zr metal at 1000°C. While this analytical method is not original, it is offered as a operating procedure for routine Zr samples. (auth)

724 KAPL-M-ERP-1
[Knolls Atomic Power Lab., Schenectady, N. Y.]
ASCORBIC ACID PROCEDURE. E. R. Proud. Dec. 3, 1947. Decl. Feb. 14, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The procedure for spectrophotometric determination of U using ascorbic acid is given. (T.R.H.)

725 LA-1953(Del.)
Los Alamos Scientific Lab., N. Mex.
DETERMINATION OF ALUMINUM IN PLUTONIUM. M. E. Smith. Aug. 1955. Decl. with deletions Mar. 14, 1957. 25p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

A spectrophotometric method is described for the determination of Al in Pu–Al alloys. The alloys are dissolved in $HClO_4$ and the diluted solution (1M) electrolyzed using a mercury cathode to remove any Fe present which would interfere in the subsequent analysis. The Pu is then separated from the Al by precipitation as the insoluble iodate, and the Al determined in the supernatant solution using aluminon as a color reagent. By proper choice of absorption cells and use of 20-mg samples, Al in alloys containing approximately 0.02 to 0.04 wt.% Al may be determined with a standard deviation of 5% or better, whereas alloys containing approximately 0.04 to 0.4 wt.% Al may be determined with a standard deviation of 3%. Reduction of the sample size is allowable for the analysis of alloys containing more than 0.4 wt.% Al. (auth)

726 TID-5201
[Manhattan District, Oak Ridge, Tenn.]
NEUTRON ABSORPTION METHODS OF ANALYSIS. L. S. Myers, Jr. [1945?]. Decl. Mar. 19, 1957. 23p. $3.30(ph OTS); $2.40(mf OTS).

The neutron-absorption tests were found to provide a means of controlling the purity of graphite and U throughout the manufacturing process. The functional test permits a direct determination of the effect that all impurities present will have on the reactor without reference to the nature of the impurities. The shotgun test was found to be less accurate because several elements are lost in the process of preparing the sample for the test, and the energy distribution of the neutrons used in the method is somewhat different from that found in the reactor. (J.E.D.)

Fluorine and Fluorine Compounds

727 CF-50-5-113
Oak Ridge National Lab., Tenn.
LITERATURE SURVEY OF DRY FLUORIDE METHODS FOR SEPARATING URANIUM, PLUTONIUM, AND FISSION

PRODUCTS. R. E. Leuze. May 11, 1950. Decl. Mar. 18, 1957. 27p. $4.80(ph OTS); $2.70(mf OTS).

728 CF-54-4-195(Del.)
[Oak Ridge National Lab., Tenn.].
DISCUSSION AND RESULTS FOR THE DIELECTRIC HEATING TEST. A. L. Southern. Apr. 27, 1954. Decl. with deletions Feb. 19, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Tests were conducted to determine the feasibility of melting mixed fluorides by dielectric heating. The results indicate fluorides become conductors as they approach the melting point. (D.E.B.)

729 CF-55-8-16(Del.)
Oak Ridge National Lab., Tenn.
THE THERMAL CONDUCTIVITY OF K_3CrF_6. H. W. Hoffman. Aug. 1, 1955. Decl. with deletions Mar. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The thermal conductivity of solid K_3CrF_6 over the temperature range 100 to 200°F was found to be 0.13 Btu/hr-ft^2-°F. (L.T.W.)

730 CF-55-8-22(Del.)
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE VISCOSITIES OF $KBeF_3$ AND $NaBeF_3$ AND SOME OBSERVATIONS ON ($LiF-BeF_2$; 50–50 Mol %). S. I. Cohen and T. N. Jones. Aug. 1, 1955. Decl. with deletions Mar. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Viscosity measurements were made on two binary salt mixtures containing 50 mole % BeF_2. The components used as cations in the mixtures were KF and NaF. The viscosity of $KBeF_3$ varied from about 20 centipoises at 570°C to about 3.3 centipoises at 800°C and may be represented throughout this temperature range by $\mu = 0.00425\exp(7096/T)$, where T is in °K. The viscosity of $NaBeF_3$ varied from about 15 centipoises at 600°C to about 5 centipoises at 800°C and may be represented throughout this temperature range by $\mu = 0.0411\exp(5148/T)$. Some observations on $LiF-BeF_2$ (50–50 mole %) are briefly reported. (L.T.W.)

Graphite

731 BNL-2506(Del.)
Brookhaven National Lab., Upton, N. Y.
PROPOSAL FOR ANNEALING OF GRAPHITE STRUCTURE IN THE BROOKHAVEN NATIONAL LABORATORY REACTOR. Sept. 4, 1953. Decl. with deletions Feb. 12, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

732 HW-24109
[General Electric Co. Hanford Atomic Products Operation, Richland, Wash.]
WATER ADSORBED BY GRAPHITE AT VARIOUS RELATIVE PRESSURES. C. N. Spalaris. July 21, 1953. Decl. Apr. 4, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80 (ph OTS); $1.80(mf OTS).

The quantity of water adsorbed by graphite at various relative pressures was determined and is presented in graphical and tabular form. (auth)

733 HW-36672(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
MODERATOR EVALUATION FOR THE K PILES—AGHT (CHF) AND 185 W GRAPHITE. G. R. Sparks and W. C. Riley. May 11, 1955. Decl. with deletions Feb. 22, 1957. 17p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40 (mf OTS).

In order to evaluate AGHT and 185-W graphites for use in the K piles, measured physical properties were directly compared to physical properties of other graphite components of the pile moderator. The results of this study indicated that neither of these graphites were significantly different in physical properties from graphites to which they were being compared. However, the exposure periods in Hanford piles for the determination of radiation induced physical property changes were short with respect to exposure periods for the evaluation of other graphites to be used as portions of the K-pile stacking. The evaluation with respect to individual physical properties is discussed in detail. (auth)

734 KAPL-1204
Massachusetts Inst. of Tech., Cambridge.
SMALL ANGLE X-RAY SCATTERING STUDY OF RADIATION DAMAGE IN GRAPHITE. B. E. Warren and D. R. Chipman. Oct. 1, 1954. Decl. Apr. 11, 1957. 23p. For Knolls Atomic Power Lab. Contract W-31-109-Eng-52, Subcontract AT(30-3)-49. $3.30(ph OTS); $2.40(mf OTS).

A small angle x-ray scattering study of damaged graphite has been made to determine whether useful information concerning radiation damage in graphite can be obtained by this technique. The results indicate that the scatterers are disk-shaped regions with the disk axis parallel to "c." The regions could be the places where interstitials spread apart the graphite layers, or they could equally well be the vacancies produced when a group of atoms has been knocked out of a layer. In either case the disk-shaped regions have a diameter of the order of 5 to 10 A, and a thickness of the order 2 or 3 A. Because of the intense small angel scattering from unirradiated graphite, the effect of radiation damage is largely obscured, and we conclude that small angle scattering studies of radiation damaged graphite give very little definite information. (auth)

735 NAA-SR-14(Del.)
North American Aviation, Inc., Downey, Calif.
RESULTS OF RESISTIVITY-RANGE MEASUREMENTS ON GRAPHITE BOMBARDED WITH CHARGED PARTICLES. F. Faris. Oct. 12, 1950. Decl. with deletions Feb. 8, 1957. 23p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

The results of resistivity-range measurements on C-18 graphite bombarded with 35.6-Mev alpha particles, 8.6-Mev protons, and 18.4-Mev deuterons from the Crocker Radiation Laboratory's 60-inch cyclotron are reported. The graphite was cut parallel to the direction of molding pressure and had a density of 1.62 gm/cm^3. The samples were oscillated during bombardment in order to avoid a sharply peaked beam profile. The quantity $((\rho/\rho_0)-1)$, in which ρ_0 and ρ are the resistivities before and after bombardment, respectively, was found to increase along the range of the particles to a maximum value followed by a decrease to zero. Saturation gives rise to a depression of the peak value relative to that at the beginning of the range as the charge density increases. The results are consistent with the view that for the conditions of the experiment (bombardment with samples below -100°C) radiation annealing effects are either small or affect the curves in the same way throughout the range. The data together with those from some bombardments with one Mev electrons indicate a value in the range from 13.4 to 17.8 ev for the energy necessary to displace a carbon atom from its lattice. When the experimental error is considered, the result is in good agreement with the accepted value of 24.7 ev. A comparison of the relative effectiveness of Hanford irradiations and cyclotron bombardments is made. At the beginning of the range it is found that 1.35 μah/cm^2 of deuterons, 1.23 μah/cm^2 of protons and 0.60

$\mu ah/cm^2$ of alpha particles are equivalent to one Mwd/T of irradiation at Hanford. (auth)

736 NAA-SR-119(Del.)
North American Avaition, Inc., Downey, Calif.
CHANGES IN THE THERMAL AND ELECTRICAL PROPERTIES OF IRRADIATED GRAPHITE DURING PULSE-ANNEALING. A. S. Hook. Jan. 24, 1952. Decl. with deletions Feb. 27, 1957. 48p. Contract AT-11-1-GEN-8. $7.80(ph OTS); $3.30(mf OTS).

An apparatus for the simultaneous experimental determination of the changes occurring in the thermal resistivity, the electrical resistivity, and the thermoelectric power of irradiated graphite as a function of pulse-annealing is described. Small specimens of irradiated graphite are heated in vacuum in a series of temperature pulses of 1-min-duration. Each succeeding pulse is to a temperature higher than its predecessor. Increments of pulsing temperature are usually 50°C, and the annealing usually terminates at 2200°C. The physical properties are determined after each pulse with the specimen at a standard equilibrium temperature. Results of the pulse-annealing of a number of specimens of varied Hanford reactor exposure histories are given. These constitute a survey to show the effects which elevated specimen temperatures during exposure have upon the nature and magnitude of radiation damage accumulated in the above named properties as compared with reactor exposure at 30°C. Elevated temperatures of exposure are shown generally to result in marked decreases in accumulated radiation damage. (auth)

737 NAA-SR-1520
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
THE EFFECT OF REACTOR IRRADIATION AT TEMPERATURES BETWEEN 400° AND 700°C ON THE THERMAL CONDUCTIVITY OF GRAPHITE. R. E. Durand and D. J. Klein. July 15, 1956. Decl. Mar. 1, 1957. 33p. Contract AT(04-3)-49. $0.40(OTS).

The thermal conductivity of AGOT-CSF graphite has been measured both during and after neutron irradiation at temperatures between 350 and 650°C. A radial heat flow measurement during irradiation in the high flux region of the MTR showed that the thermal conductivity saturated at a value greater than one-third the initial value; this was borne out by post-irradiation measurements made in the laboratory. The graphite moderator of the SGR should therefore not cause heat-transfer problems due to deterioration of its conductive properties, since it will operate in the temperature range investigated and in a similar neutron flux. (auth)

738 TID-10001
Battelle Memorial Inst., Columbus, Ohio.
STUDIES OF GRAPHITE FOR FUEL ELEMENTS. L. D. Loch, J. A. Slyh, and W. H. Duckworth. Oct. 13, 1954. Decl. Mar. 16, 1957. 67p. Contract W-7405-eng-92. $0.45(OTS).

Improved graphites for use in fuel elements were developed from artificial graphites and phenolic resins. These materials are particularly well suited for fuel-element preparation. The resins do not soften during baking, so bodies hold their shape and show good dimensional uniformity. The physical properties of graphite were obtained without heating at "graphitizing" temperatures, which would be detrimental to UO_2 or other inclusions of nuclear fuel. High densities and bending strengths, viz., 1.80 g/cm^3 and 15,000 psi, were obtained directly without the usual impregnation of the baked product with pitch and rebaking. At 2770°F, one of the experimental bodies had a tensile strength of 8250 psi as compared to 4700 psi for Type ECA, a commercial graphite of interest for fuel elements. UO_2-bearing bodies were prepared by using the improved graphites as matrices. The UO_2 was added during primary fabrication of the bodies by several different methods which produced a wide range of particle sizes. Consideration of the qualitative effects of UO_2 particle size and of some of the experimental results indicates that the use of UO_2 nodules between 100 and 800 microns in diameter may have three important advantages: (1) less fission damage to the graphite, (2) less diffusion and loss of U, and (3) better retention of fission products by the fuel element. (auth)

Laboratories and Equipment

739 HW-17769(Del.)
Hanford Works, Richland, Wash.
TERMINAL REPORT ON THE RADIOLANTHANUM LABORATORY. I. LABORATORY AND LABORATORY EQUIPMENT DESIGN. J. K. Figenshau. July 21, 1950. Decl. with deletions Feb. 19, 1957. 140p. Contract [W-31-109-eng-52]. $21.30(ph OTS); $6.90(mf OTS).

The status of the planning of the analytical control laboratory for the radiolanthanum (RaLa) process at its termination was summarized. The design of laboratory apparatus and equipment is included. (F.S.)

Radiation Effects

740 CF-56-8-188
Oak Ridge National Lab., Tenn.
CORROSION OF PLATINUM BY UO_2SO_4 SOLUTIONS UNDER IRRADIATION. G. H. Jenks. Aug. 30, 1956. Decl. Mar. 19, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Results indicate that corrosion of the metal is accelerated to a small extent during irradiation. A summary of the results from in-pile loops and in-pile autoclaves is given. (L.M.T.)

741 DC-51-10-39(Del.)
[General Electric Co. Aircraft Nuclear Propulsion Project, Cincinnati].
RADIATION DECOMPOSITION OF WATER MODERATOR: SURVEY OF EXISTING INFORMATION. Task No. 7501. F. H. Welch. Oct. 15, 1951. Decl. Mar. 9, 1957. 17p. $3.30(ph OTS); $2.40(mf OTS).

A survey of the available information on the radiation damage to water is presented. (auth)

742 KAPL-M-CWT-7
Knolls Atomic Power Lab., Schenectady, N. Y.
RETENTION OF HELIUM BY NEUTRON IRRADIATED HOT-PRESSED BORON CARBIDE. Charles W. Tucker, Jr. Oct. 18, 1954. Decl. Mar. 7, 1957. 12p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Measurements of the B-11/B-10 ratio, helium evolution, and Li-6/Li-7 ratio show that the burnup in a disc of hot pressed boron carbide is of the order of 1% of all atoms present. These data are not sufficiently precise to determine whether any helium escaped the sample during irradiation. However, the helium evolution experiments show that the helium left is held very strongly by the crystalline lattice to temperatures as high as 1100°C and that the major portion of the gases are not released until a temperature of 2000°C (nearly the melting point) is reached. (auth)

743 TID-5148(Del.)
California Research Corp., Richmond, Calif.
RESEARCH ON THE RADIATION STABILITY OF ORGANIC FLUIDS. Progress Report [for] January 1, 1953 through

PRODUCTS. R. E. Leuze. May 11, 1950. Decl. Mar. 18, 1957. 27p. $4.80(ph OTS); $2.70(mf OTS).

728 CF-54-4-195(Del.)
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Graphite

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743 TID-5148(Del.)
California Research Corp., Richmond, Calif.
RESEARCH ON THE RADIATION STABILITY OF ORGANIC FLUIDS. Progress Report [for] January 1, 1953 through

September 30, 1953. Oct. 30, 1953. Decl. with deletions Feb. 7, 1957. 105p. Contract AT(11-1)-174, Report No. 6. $16.80(ph OTS); $5.70(mf OTS).

Additional knowledge was gained pertaining to effects of irradiation variables, chemical components, etc., on the stability of organics. This basic program was supplemented with an applied program wherein research was undertaken of particular interest in reactor moderator-coolant and lubricant applications. Irradiations of the capsule type in the absence of air were conducted in the MTR, BNL, and X-10 reactors under a variety of conditions in investigations of effects of temperature, flux, dosage, and organic chemical structure. The MTR gamma facility and the California Research Co^{60} source were also used. With highly aromatic organics, damage measured by viscosity change was found independent of irradiation temperature up to 371°C. Coke formation was minor, although it was universal at 426°C. Many fused ring and nonfused ring aromatics were stable at 371°C. Viscosity changes were slight, and gassing amounted to about 2 to 5 ml gas/ml fluid. Simple molecular distillations showed biphenyl, naphthalene, and 1,3-diphenylbenzene, to be 93 to 97% distillable, of which 72 to 81% was pure starting material. The stabilities of solutions of highly viscous polymers in low viscosity alkylbenzenes were studied in the presence of air with gamma radiation. Styrene-alkene copolymer, polyalkene, and polyester solutions markedly decreased in viscosity on irradiation; whereas, polystyrene solutions merely increased slowly. Comparative stabilities of representative types of organics with combined neutron and gamma radiation allowed the tentative determination, based on viscosity change, that one fast neutron (>1 Mev) causes ~16 times the damage of one gamma (1.6 Mev). Thermal neutrons cause insignificant damage. As a result, it is suggested that the radiation dosage to cause damage be expressed in terms of "equivalent fast neutrons," a number obtained by adding to the number of fast neutrons the damage equivalent number each of gamma rays and epithermal neutrons. (auth)

Rare Earths and Rare-earth Compounds

744 CN-2010

Clinton Labs., Oak Ridge, Tenn.

THE FLOCCULATION OF LANTHANUM FLUORIDE. Final Report [for] April 1, 1944 to July 15, 1944 [on] Problem Assignment No. 211-X4AS. J. H. Balthis. Nov. 8, 1944. Decl. Feb. 14, 1957. 29p. $4.80(ph OTS); $2.70(mf OTS).

745 KAPL-1271

Knolls Atomic Power Lab., Schenectady, N. Y.

IMPLICATIONS OF THE U-232 DECAY CHAIN ACTIVITY FOR THE PROCESSING OF IRRADIATED THORIUM AND RECOVERED U-233. R. C. Feber. Feb. 4, 1955. Decl. Apr. 3, 1957. 32p. Contract W-31-109-eng-52. $6.30 (ph OTS); $3.00(mf OTS).

The problems introduced by the associated beta and gamma radiation from the U^{232} decay chain in U^{233} processing are of sufficient magnitude to introduce new considerations with respect to handling the U^{233} product, processing and recycling the irradiated thorium, and setting decontamination specifications. The U^{233} from an MTR slug had a U^{232} content which would result in a γ-ray dosage of about 600 r/hr at the surface of a 1 kg sphere of U^{233} after secular equilibrium was reached in ten years. More than half as much radiation would be present at the end of two years. Remote handling of such U^{233} would be required at even shorter periods after isolation. If this high U^{232} content cannot be tolerated, much lower g/ton levels must be used, since the fraction of the U^{232} in U^{233} is roughly proportional to the g/ton level. The bulk of the β- and γ-emitting daughters of U^{232} which are associated with the separated Th grows in during the cooling period prior to processing. A 4-in.-diameter billet of thorium 10 in. in length recovered from the MTR irradiated slug which had been cooled 200 days would have a γ-ray dosage of 1.1 r/hour at the surface of the cylinder. This radiation would decay with the 1.9 year half life of Th^{228}. However, if the thorium were infinitely recycled with 270 days irradiation at the MTR flux, 200 days cooling and 100 days allowed for slug refabrication, the dose rate at the surface of the billet would reach a limiting value of 2.1 r/hour. This would require remote handling facilities for slug refabrication procedures. The β and γ activities of the U^{232} decay chain must be taken into account in setting decontamination specifications for Th. (auth)

Separation Procedures

746 ANL-4329(Del.)

Argonne National Lab., Lemont, Ill.

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR APRIL, MAY, AND JUNE, 1949. Leslie Burris, Jr., comp. Decl. with deletions Feb. 12, 1957. 66p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

Progress is reported in developmental studies on the Redox Process and studies on the chemistry of Ru. (C.H.)

747 ANL-4372(Del.)

Argonne National Lab., Lemont, Ill.

CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR JULY, AUGUST, AND SEPTEMBER, 1949. Leslie Burris, Jr., comp. Decl. with deletions Feb. 12, 1957. 72p. Contract W-31-109-eng-38. $12.30(ph OTS); $4.50 (mf OTS).

Redox Process Studies—Plutonium Processing. Laboratory batch extraction of second Pu cycle feed revealed an unextractable portion of Pu. The amount of unextractable Pu could be reduced by stronger oxidation with $Na_2Cr_2O_7$, O_3, or $(NH_4)_2S_2O_8$, however decontamination from gamma activity was reduced. Precipitation of Pu which occurred in these runs, was prevented by hexone removal and the precipitate dissolved in several oxidizing solutions. Removal of Ru by Ozonization. An ozonization tower was installed and one active run completed. Uranium Extraction with TBP. Lab batch extraction runs to determine decontamination factors and to compare TBP with hexone are reported. Neptunium Chemistry. The solvent extraction behavior of Np was found to be dependent on acidity, Np^{6+} not extracting into hexone in the acid-deficient Redox flowsheet. In the IB column, Np^{4+} maybe directed with U or Pu by slight changes in acidity. Mixer-Settler Extractors. Units were installed and a test run was completed. Ruthenium Chemistry. The reaction between NO_2^- ion and $Ru(NO_3)_4$ solutions was studied as affected by temperature. Nitric oxide gas was found to produce spectra similar those of NO_2^- addition experiments. Polarography studies of Ru^{4+} in HCl, HNO_3, H_2SO_4, and $HClO_4$ were completed. Waste Processing. Lab work on a procedure for simultaneous recovery and decontamination of $Al(NO_3)_3$ from Redox streams is described. A crystallization and recrystallization method was studied. Production of Hafnium-Free Zirconium. A distillation column was set up and Zr (< 0.2% Hf) was distilled from $POCl_3$ complexes. (T.R.H.)

748 ANL-5176(Pt. 6(Del.))

Division of Reactor Development, AEC and Argonne National Lab., Lemont, Ill.

REACTOR INFORMATION MEETING, ARGONNE NATIONAL LABORATORY, OCTOBER 7, 8, 9, 1953. PAPERS. PART 6. PROCESSING. Decl. with deletions Feb. 11, 1957. 36p. $6.30(ph OTS); $3.00(mf OTS).

The following papers are presented: design of the Idaho Chemical Processing Plant; recovery methods for submarine reactor fuels; LMFR processing loops; present status of recovery in the fluoride volatility process; and adaptation of the fluoride volatility process to various reactor fuels. (L.M.T.)

749 ANL-5529(Del.)
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER, 1955. Feb. 10, 1956. Decl. with deletions Feb. 12, 1957. 130p. Contract W-31-109-eng-38. $19.80(ph OTS); $6.30(mf OTS).

Progress is reported on the following investigations: direct-cycle boiling-reactor entrainment studies; theory of solvent extraction; fused-salt process for U recovery from STR-type fuel elements; melt refining of EBR fuel elements; volatilization of PuF_6 from Pu–U alloys, and x-ray spectrometric determination of U. (C.W.H.)

750 ANL-LB-SL-1016
Argonne National Lab., Lemont, Ill.
REVISED EBR-II FLOWSHEET FOR FUEL CYCLE WITH FUEL PURIFICATION BY OXIDATIVE SLAGGING. L. Burris, Jr. and I. G. Dillon. Feb. 23, 1956. Decl. Mar. 11, 1957. 15p. Contract [W-31-109-eng-38]. $3.30(ph OTS); $2.40(mf OTS).

751 ANL-ML-SL-1025
[Argonne National Lab., Lemont, Ill.]
PYRO-MET PLUTONIUM FLOWSHEET. M. Levenson. Mar. 5, 1956. Decl. Mar. 13, 1957. 2p. Contract [W-31-109-eng-38]. $1.80(ph OTS); $1.80(mf OTS).

A preliminary flowsheet for a pyrometallurgical process for Pu breeder cores is given. (T.R.H.)

752 CC-739
Chicago. Univ. Metallurgical Lab.
CHEMICAL RESEARCH — U^{233} PRODUCTION AND EXTRACTION: REPORT FOR MONTH ENDING JUNE 21, 1943. Decl. Feb. 15, 1957. 12p. Contract [W-7401-eng-37]. (A-948). $3.30(ph OTS); $2.40(mf OTS).

753 CF-45-2-9
Clinton Labs., Oak Ridge, Tenn.
TECHNICAL DIVISION SECTION I REPORT FOR THE WEEK ENDING FEBRUARY 4, 1945. M. D. Peterson. Decl. Feb. 16, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

754 CF-51-8-16(Del.)
Oak Ridge National Lab., Tenn.
AN INTRODUCTION TO RADIOCHEMICAL SEPARATIONS TECHNOLOGY: LECTURE SERIES. (A Summary of Lecture Notes Prepared by du Pont Personnel, March 23, 26, 27, 28, and 29, 1951.) E. J. Weber, ed. Aug. 24, 1951. Decl. with deletions Feb. 14, 1957. 87p. Contract W-7405-eng-26. $13.80(ph OTS); $4.80(mf OTS).

Included are brief lecture notes on the protection of personnel from radiation, a survey and theory of nuclear reactors, a survey of various separation processes, radiochemical analyses, separation of radioisotopes, U mining and metallurgy, and the gaseous diffusion process. (W.L.H.)

755 CF-52-7-78(Chaps. XVII, XVIII, XIX, XXXII, XXXIV, XXXV)(Del.))
Oak Ridge School of Reactor Technology, Tenn.
CHEMISTRY COURSE SYLLABUS. Sigfred Peterson, W. K. Eister, and R. W. Stoughton, comps. and eds. CHAPTERS XVII, XVIII, XIX, XXXII, XXXIV, XXXV, [on] PRECIPITATION PROCESSES, ANALYTICAL CHEMISTRY AND ORE PROCESSING. R. W. Stoughton, I. Perlman, P. F. Thomason, K. B. Brown, and M. W. Gerrard. Jan. 12, 1953. Decl. with deletions Feb. 13, 1957. 93p. $15.30(ph OTS); $5.40(mf OTS).

The textbook continues with chapters on chemical separation principles, precipitation processes for Pu, other precipitation processes, ionic analytical chemistry, U ores and processing, and Th ores and processing. (T.R.H.)

756 CF-53-10-113
Oak Ridge National Lab., Tenn.
DESIGN STUDY NO. 2 FOR ALTERNATE SLUG HANDLING AND CHARGING FOR THE THOREX PROCESS PILOT PLANT. F. A. Heddleson. Oct. 16, 1953. Decl. Apr. 8, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The design and operation of a combination slug carrier and integral feeder with a single discharge chute are given. (D.E.B.)

757 CF-53-12-150(Pt.1(Del.2))
Oak Ridge School of Reactor Technology, Tenn.
CHEMICAL SEPARATIONS PROCESSES. PART 1. (CHAPTERS 1–8, 10, AND 11). Sigfred Peterson, comp. and ed. Dec. 1953. Decl. with deletions Mar. 20, 1957. 108p. $16.80(ph OTS); $5.70(mf OTS).

All the separation processes conceived and investigated are presented and described with only enough detail to illustrate. Solvent extraction, precipitation, sorption, volatility, electrochemical, and high temperature processes are treated along with special dissolution processes, processes for irradiated fuels, and U recovery from waste solutions. (T.R.H.)

758 CF-54-7-154
Oak Ridge National Lab., Tenn.
SUGGESTED SEPARATION FACTOR FOR CALCULATIONS ON PLUTONIUM ISOTOPE SEPARATION BY ION EXCHANGE RESINS. G. H. Clewett. July 26, 1954. Decl. Feb. 15, 1957. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

759 CF-54-10-145(Del.)
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR PERIOD ENDING OCTOBER 15, 1954. R. E. Blanco and D. E. Ferguson. Oct. 25, 1954. Decl. with deletions Feb. 14, 1957. 11p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

760 CF-54-12-55
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROCESSING: RECOMMENDATION FOR THE TBR CHEMICAL PROCESSING HYDROCLONE SYSTEM. C. E. Guthrie. Dec. 3, 1954. Decl. Apr. 4, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Six hydroclone systems were investigated from the standpoint of concentration factor, power requirements, and number of flow controls required for operation. The operating characteristics of three systems with a varying ratio of overflow to underflow of 9/1 and larger were developed. (F.S.)

761 CF-55-3-40(Del.)
Oak Ridge National Lab., Tenn.
THE EFFECT OF SIZE OF THOREX PLANT ON PROCESSING COSTS ASSOCIATED WITH A THERMAL BREEDER REACTOR POWER STATION. E. D. Arnold. Mar. 3, 1955. Decl. with deletions May 9, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

762 CF-55-7-42
Oak Ridge National Lab., Tenn.
HRT SOLIDS PRODUCTION RATE CONSIDERATIONS IN CHEMICAL FACILITY DESIGN. H. E. Williamson. July 12, 1955. Decl. Apr. 17, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The effects of solids production rate on the design of the chemical facility for the HRT are discussed. It is necessary to know the degree to which solids can be concentrated in the evaporator without limiting operation or rendering a slurry too difficult to handle. Additional decay storage capacity in the chemical process cell seems desirable. (F.S.)

763 CF-55-10-124
Oak Ridge National Lab., Tenn.
THOREX PILOT PLANT: EVALUATION OF EQUIPMENT PERFORMANCE FOR THE PERIOD OCTOBER 1954 THROUGH MARCH 1955. A. J. Pecquet, W. R. Winsbro, and G. S. Sadowski. Oct. 14, 1955. Decl. Feb. 16, 1957. 58p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60 (mf OTS).

Descriptions and performances of the mechanical equipment (pulse columns, pulse generators, pumps, etc.) used in the Thorex Process Pilot Plant are presented. Information is also presented on the case of direct maintenance of unit-shielded equipment. (C.W.H.)

764 CF-56-5-19(Del.)
Oak Ridge National Lab., Tenn.
FUSED SALT PROCESSING: TENTATIVE FLOWSHEET. R. G. Wymer. May 1, 1956. Decl. with deletions Mar. 13, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A tentative flowsheet for fused fluoride salt dissolution of U–Zr fuel material is presented. (T.R.H.)

765 CF-56-7-65(Del.)
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING JUNE 29, 1956. R. E. Blanco and D. E. Ferguson. July 11, 1956. Decl. with deletions Mar. 13, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

766 CF-56-8-138
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT, SECTIONS A AND B [FOR] WEEK ENDING AUGUST 10, 1956. R. E. Blanco and D. E. Ferguson. Aug. 17, 1956. Decl. Mar. 19, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

767 CF-56-8-190
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT, SECTIONS A AND B [FOR] WEEK ENDING AUGUST 17, 1956. R. E. Blanco and D. E. Ferguson. Aug. 23, 1956. Decl. Mar. 12, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

768 CN-1017(Del.)
Chicago. Univ. Metallurgical Lab.
SURVEY OF SEPARATION PROCESSES. PART I. GENERAL CONCLUSIONS AND COMPARISONS. W. W. Armstrong, E. R. Gilbert, and J. O. Maloney. Oct. 29, 1943. Decl. with deletions Feb. 15, 1957. 27p. $6.30(ph OTS); $3.00(mf OTS).

The Bismuth Phosphate, Lanthanum Fluoride, Adsorption-Sodium Uranyl Acetate, and Dry Fluoride Processes were surveyed for their efficiency in recovering Pu in a decontaminated form. The Bismuth Phosphate Process offered the best results but it was felt that the Lanthanum Fluoride process could be developed. The Sodium Uranyl Acetate Process was extremely unsatisfactory and the Dry Fluoride Process was found to require extensive experimental work before it could be evaluated. (F.S.)

769 CN-1373
[Chicago. Univ.] Metallurgical Lab.
ABSORPTION METHODS FOR THE EXTRACTION AND DECONTAMINATION OF PLUTONIUM. R. H. Beaton, P. R. Fields, T. J. LaChapelle, D. C. Lincoln, and I. Sheft. Feb. 23, 1944. Decl. Apr. 4, 1957. 37p. (A-2016). $6.30 (ph OTS); $3.00(mf OTS).

Investigation of a two-cycle, all-adsorption, extraction and decontamination process for Pu continued. Studies directed at producing higher decontamination in the Amberlite IR-1 extraction cycle are described along with preliminary work on the coupling of the extraction cycle to the $BiPO_4$ decontamination cycle. (T.R.H.)

770 CN-2223
[Clinton Labs., Oak Ridge, Tenn.]
THE AGING OF BISMUTH PHOSPHATE PRECIPITATES. Final Report [for] Period July 15, 1944 to January 1, 1945 [on] Problem Assignment No. 203-X72C. Part II. Arthur W. Adamson. Jan. 15, 1945. Decl. Apr. 4, 1957. 41p. Contract W-7405-eng-39. $12.30(ph OTS); $4.50 (mf OTS).

Studies on the changes that occur in by-product $BiPO_4$ precipitates which are responsible for the incorporation of foreign activities during digestion revealed the importance of dynamic recrystallization as a mechanism for the uptake or for self-purification. A quantitative theory led to the equation $\ln (A-A\infty/A_0) = \ln (K/K + m) - k_2S(K + m)t$ where A and A∞ denote the concentration in solution of the species taken up by the precipitate at time t and at equilibrium; A_0, the total concentration of A in the system; K, the mass law (Nernst) distribution constant; m, the ratio of the total amount of Bi^{3+} in solution to that in the solid; k_2, the rate of solution constant; and S the total surface area of the precipitate. Hence, in a given situation where m is arbitrarily fixed, (and the composition of the solid does not change) it is possible to estimate the amount of an isomorphous fission product ion present in a $BiPO_4$ precipitate after any time up to 6 hours if S, k_2 and K are known. In solutions where the $BiPO_4$ solubility is very low, it was found that the pick-up of activity could be described quantitatively on the basis of diffusion through the solid. Using the mathematical solution for the radial diffusion equation, a value of $D = 3 \times 10^{-17}$ cm^2/min at 75° for RaE in $BiPO_4$ was computed. A general conclusion from the experiments reported is that such aging processes as crystal self-perfection, agglomeration, irreversible cementing of particles, and polymorphic transitions are not important or becomes unimportant within a few minutes after precipitation. (auth)

771 HW-12836(Del.)
Hanford Works, Richland, Wash.
THE SOLVENT-EXTRACTION PERFORMANCE OF SMALL SCALE REDOX IA COLUMNS USING FENSKE HELIX PACKING, AMMONIUM NITRATE SALTING AGENT AND UNIRRADIATED URANIUM. REDOX TECHNICAL DATA STUDY NO. 12. A Semi-Works Progress Report. F. W. Woodfield, J. G. Bradley, W. O. Switzer, and E. T. Merrill. Apr. 13, 1949. Decl. with deletions Feb. 17, 1957. 25p. $6.30(ph OTS); $3.00(mf OTS).

Redox IA Column studies made in small-scale columns packed with $^3/_{16}$-in. Fenske helices and using NH_4NO_3 salting agent are presented. Preliminary IA Column studies were conducted in Pyrex glass columns varying in

diameter from ½ to 2 inches. Following a shutdown for equipment revisions intended to increase the limiting capacity, and to facilitate obtaining design data for a 1/25 scale Redox Test Plant, a second series of IA Column studies were conducted in Pyrex glass columns varying in diameter from 1 to 3 inches. These studies were conducted to procure performance data for uranium transfer, to determine limiting volume velocities under Redox IA Column conditions (A.N.L. NH_4NO_3 Flowsheet), and to define certain features of the packing performance. It was found that the initial design did not permit smooth operation of the equipment at the volume velocities stipulated by the A.N.L. NH_4NO_3 Flowsheet (178 gal/(hr)(sq. ft.)). It was found possible to operate up to the complete flooding capacity which was about 150% of the specified Flowsheet rates but local floods which formed and dissipated on a somewhat regular schedule created very unsteady conditions in the columns at any volume velocity above 55 to 75% of the Flowsheet value. For this reason the most reliable H.E.T.S. (Height Equivalent to a Theoretical Stage) values were obtained at 50 to 60% of Flowsheet volume velocities. The pertinent uranium H.E.T.S. values are summarized below. The effect of feed source is defined in the tabulation. (auth)

772 HW-14532
Hanford Works, Richland, Wash.
OBSERVATIONS AND EVALUATION OF THE DECONTAMINATION PROCEDURES EMPLOYED AT OAK RIDGE NATIONAL LABORATORIES REDOX PILOT PLANT EQUIPMENT. J. M. Smith, Jr. and J. G. Bradley. Sept. 19, 1949. Decl. Feb. 19, 1957. 14p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The ORNL Pilot Plant, consisting of stainless vessels, reactors, and piping assembled by bolted, gasketed flanges and housed in a monolithic concrete structure concluded a series of Redox runs employing Hanford-level irradiated U. The process equipment was cleaned sufficiently by chemical and physical methods to allow direct maintenance with reasonable working time limits. The dissolver, which should be the most difficult vessel to decontaminate, gave a maximum internal surface dosage rate of less than 50 mrep/hr following decontamination. Unprotected concrete and structural iron surfaces in Cell 1 were not sufficiently decontaminated during approximately 3 weeks of cleaning. It is believed that by a proper choice of decontaminating solutions together with improved order and conditions of application, the efficiency of decontamination could be improved and time cycles and material consumption reduced. (auth)

773 HW-17179
Hanford Works, Richland, Wash.
EQUILIBRIUM RELATIONSHIPS FOR STRIPPING URANIUM FROM SOLUTIONS OF TRIBUTYL PHOSPHATE AND HYDROCARBON DILUENT. Fred Clagett. Mar. 1, 1950. Decl. Feb. 19, 1957. 11p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

774 HW-17747
Hanford Works, Richland, Wash.
URANIUM PHASE EQUILIBRIA IN THE TBP PROCESS. D. P. Granquist and E. T. Merrill. March 1, 1951. Decl. Feb. 19, 1957. 39p. Contract W-31-109-Eng-52. $7.80 (ph OTS); $3.30(mf OTS).

Correlations have been prepared for U equilibria in RA and RC type columns and are presented. Correlation of RC equilibrium data was much simpler than that of RA equilibrium data since in the RC system HNO_3 and UNH are the only solutes present in significant amounts, while the RA system also contains H_3PO_4, H_2SO_4, $NaNO_3$ and ferrous sulfamate. The final equilibrium correlations for TBP-HW #4 Flowsheet conditions indicate that approximately 3.5 equivalent stages are required in both the RA Column extraction section and the RC Column to reduce U losses below 0.5% of the feed U in the waste stream from each column. The corresponding number of transfer units is approximately 7.0 for both columns. (auth)

775 HW-33479(Rev.)
Hanford Works, Richland, Wash.
RUTHENIUM IN THE PUREX PROCESS. R. E. Tomlinson and F. J. Leitz, Jr. Oct. 14, 1954. Decl. Mar. 7, 1957. 6p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80 (mf OTS).

776 ISC-141
Ames Lab., Ames, Iowa.
PROGRESS REPORT IN CHEMICAL ENGINEERING FOR THE PERIOD OCTOBER 1, 1950 TO MARCH 31, 1951. G. L. Bridger, G. H. Beyer and W. R. Millard. May 14, 1951. Decl. Mar. 27, 1957. 25p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported on production of Th from monazite, separation of Zr compounds from Hf compounds, production of stable slurries for homogeneous reactor use, recovery of U from phosphate ores, and fundamental studies of liquid-liquid extraction. A detailed progress report on two processes for the separation of Zr and Hf is presented namely, (1) silica gel adsorption and (2) liquid-liquid extraction, using tributyl phosphate-butyl ether solvent. (auth)

777 ISC-393(Del.)
Ames Lab., Ames, Iowa
QUARTERLY SUMMARY RESEARCH REPORT IN ENGINEERING FOR THE PERIOD APRIL, MAY, AND JUNE 1953. Aug. 20, 1953. Decl. with deletions Feb. 26, 1957. 11p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40 (mf OTS).

Progress of the work is briefly reported on the production of Th from monazite, purification of rare earths, recovery of spent silica gel from adsorption separation of Hf and Zr, Zr process development, and studies in liquid-liquid extraction. (J.E.D.)

778 KAPL-367
Knolls Atomic Power Lab., Schenectady, N. Y.
LARGE SCALE OZONE VOLATILIZATION OF RUTHENIUM. R. C. Feber, L. W. Niedrach, and E. L. Zebroski. July 11, 1950. Decl. Mar. 28, 1957. 17p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The volatilization of ruthenium with ozone has been tested on a 50-gal scale with inactive ruthenium. Decontamination factors of 23, 22, and 49 were obtained in three successive runs. With several equipment modifications to minimize reflux and improve gas distribution, it is believed that preliminary confirmation of the design basis adequate to justify tests with active solutions has been obtained. (auth)

779 KAPL-M-Redox-1
Knolls Atomic Power Lab., Schenectady, N. Y.
REDOX MONTHLY REPORT FOR SEPTEMBER 1948. Sept. 27, 1948. Decl. Feb. 28, 1957. 25p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported on heavy element and fission product chemistry; Redox solution conductivity; analytical methods for Redox; plutonium valence states; and decontamination of Redox waste solutions. (F.S.)

780 KAPL-M-SSJ-3
Knolls Atomic Power Lab., Schenectady, N. Y.
RADIATION EFFECTS IN THE CHEMICAL PROCESSING OF FAST OXIDE BREEDER FUEL. Samuel S. Jones.

May 31, 1956. Decl. Mar. 12, 1957. 19p. Contract W-31-109-Eng 52. $3.30(ph OTS); $2.40(mf OTS).

In processing spent fuel from the KAPL Plutonium Power Breeder Reactor in the Purex Process the effect of radiation damage becomes prominent. The effect of the radiation on the following process factors is discussed: temperature, gas evolution, viscosity, corrosion, solute oxidation states, solvent deterioration, and phase interface behavior. (T.R.H.)

781 KLX-1225
Vitro Corp. of America, New York.
NITRATE RECOVERY FROM RAFFINATES. Project (23-G) Summation [for] Period October 1950—February 1952. Job 23. May 28, 1952. Decl. Mar. 28, 1957. 29p. Contract AT(30-1)-848. $0.35(OTS).

Investigations have been made on the recovery of nitrates from Mallinckrodt raffinates by liquid-liquid solvent extraction, distillation after acidification with H_2SO_4, and thermal decomposition. A thermal decomposition process has been demonstrated for the recovery of 95% of the available nitrates as HNO_3. (auth)

782 M-1408
Columbia Univ., New York. Div. of War Research.
SELF-WETTING STEDMAN PACKING. W. Kilpatrick. Feb. 15, 1945. Decl. Mar. 20, 1957. 11p. Contract W-7405-Eng-50. (100B-R-301). $3.30(ph OTS); $2.40(mf OTS).

Stedman packings were coated with nickel powders in an effort to increase distillation efficiency. The coating technique is described, and the performance of the coated packings is tabulated. (D.E.B.)

783 MCW-17
Mallinckrodt Chemical Works, St. Louis.
THE DISTRIBUTION OF PRECIOUS METALS IN VARIOUS RESIDUES OBTAINED IN THE PREPARATION OF URANYL NITRATE FROM PITCHBLENDE. R. W. Shearer. Apr. 1, 1946. Decl. Mar. 30, 1957. 9p. $0.30(OTS).

A study was made of the distribution of Ag, Au, Pt, and Pd in the residues from an extraction process in which purified $UO_2(NO_3)_2$ is prepared from a pitchblende ore. The residues studied were the gangue Pb cake, the $BaSO_4$ cake, the filtrate boildown cake, the raffinate cake, and the purified $UO_2(NO_3)_2$. Samples representing three different batches of ore were used. The following conclusions were reached: (1) neither Ag nor Pd are present in the ore in significant quantities; (2) approximately 30 ppm of Au (based on U content) is present, of which 90% was found in the residue (gangue Pb cake) from the initial co-precipitation step; this residue contains gangue, $PbSO_4$, $RaSO_4$, and other insoluble sulfates; (3) the ore contains 15 ppm of Pt, which is distributed in an extremely varying manner in the residues leaving the process. (W.L.H.)

784 MCW-19
Mallinckrodt Chemical Works, St. Louis.
DEVELOPMENT OF CONTINUOUS ETHER EXTRACTION PROCESS FOR PURIFICATION OF URANIUM DERIVED DIRECTLY FROM PITCHBLENDE ORE. J. V. Opie. Apr. 1, 1946. Decl. Feb. 26, 1957. 50p. Contract W-7405-eng-1. $0.35(OTS).

In the production of pure U compounds using pitchblende ore as a source material, a double extraction process has been developed to carry out the production in a continuous operation. The acid feed was extracted with acid ether, the resulting partially purified uranyl nitrate was concentrated and the excess acid removed. The U was then extracted in the neutral state using neutral ether. The extract was washed to remove remaining traces of impurities and then processed to produce pure UO_3. The necessary equipment, operating conditions, and reagents required are described and evaluated. (W.L.H.)

785 MCW-226
Mallinckrodt Chemical Works, St. Louis.
LABORATORY STUDIES ON THE USE OF COLORADO PLATEAU SODA SALTS (V-14 AND V-20) AS FEED MATERIALS FOR THE MCW PLANT 6 [URANIUM ORE REFINERY] PROCESS. PART I. NITRIC ACID DIGESTION. PART II. ETHER EXTRACTION. C. W. Kuhlman. Oct. 10, 1949. Decl. Feb. 27, 1957. 15p. $3.30(ph OTS); $2.40(mf OTS).

Preliminary studies indicate that feed materials containing in excess of about ⅓ gram of V per 100 grams of U will, in fully oxidized condition, result in a precipitation of uranyl vanadate during HNO_3 digestion, according to Plant 6 procedures. It is also indicated, however, that a precipitate will be dropped upon sparging in the ether house (in the NOK liquor preparation step) and that this precipitation will reduce the V content of the liquor from 3300 to about 130 ppm. Considerable data on the chemistry of V under Plant 6 conditions are given. (D.E.B.)

786 MonN-96
[Clinton Labs., Oak Ridge, Tenn.]
SEPARATION OF URANIUM(VI) FROM THORIUM USING ORGANIC COMPLEX-FORMING REAGENTS AND SOLVENT EXTRACTION. Final Report on Problem Assignment No. CX3-6 for Period June 20, 1945 to April 1, 1946. O. K. Neville. June 6, 1946. Decl. Apr. 10, 1957. 28p. Contract W-35-058-eng-71. $4.80(ph OTS); $2.70(mf OTS).

The extraction of small amounts of U from high concentrations of Th by use of organic reagents which form complexes with U and by solvent extraction of the U-organic complexes was investigated. Quantitative comparison of the degrees of extraction of U from solutions 1 M in Th and 0.001 M in U at a pH of 3 which were obtained with twenty-eight organic reagents and with hexone as a solvent showed potassium ethylxanthate and sodium diethyldithiocarbamate to be most efficient. Equal or better separation was achieved with these reagents in similar experiments in which the concentration of U was only 2.78×10^{-5} M. Systematic studies of the effect of changing the R groups in the xanthate, ROCSSK, and in the dithiocarbamate, R′R″NCSSNa, were made. The relative efficiencies of each of these compounds using various organic solvents were determined. In the case of the xanthates, the ethyl, isopropyl, n-hexyl, and cyclohexyl produced the highest separation. The best solvents were dibutyl cellosolve and dibutyl carbitol. Of the dithiocarbamates, those derived from secondary amines, especially from secondary phenylamines, were most efficient in extracting uranium. As a solvent with the dithiocarbamates, ethyl acetate gave higher U extraction and lower Th extraction than did hexone. On the basis of stability in solution, the dithiocarbamates were preferable to the xanthates. The instability of the dithiocarbamates was such, however, that extractions had to be completed within thirty minutes. However, it might be possible to extend this period by repeated addition of the reagent or by development of more stable dithiocarbamates. Under present conditions, the instability is the only serious limitation to the process. The use of dithiocarbamates, in particular of sodium ethylphenyldithiocarbamate, in a batch extraction provided more efficient U extraction than is obtainable by salting-out batch extraction methods. Advantages over the latter method include the necessity for fewer extractions and elimination of inorganic salting-out agents which might interfere with Th recovery. (auth)

787 NAA-SR-943(Rev.)
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1953. E. E. Motta, D. W. Bareis, and D. D. Cubicciotti, eds. Mar. 26, 1954. Decl. Mar. 25, 1957. 21p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

Work has continued on high temperature methods for processing irradiated U fuel. Additional results have been obtained with fused halide treatment, solid scavengers, and direct Pu distillation. With fused fluorides about 95% of the Pu was removed from a U sample, while treatment of U with HCl gas removed almost all the Pu and many fission products. Treatment of molten U with UO_2 removed a substantial fraction of the fission products without removing Pu. Uranium carbide treatment results were similar to the oxide but not as effective. A small scale distillation of Pu from U showed that Raoult's law is obeyed. (auth)

788 NAA-SR-1324
North American Aviation, Inc., Downey, Calif.
PYROCHEMICAL SEPARATIONS METHODS: II. THE DISTRIBUTION OF PLUTONIUM AND FISSION PRODUCTS BETWEEN URANIUM AND MAGNESIUM. Rex A. Barney and Francis J. Keneshea, Jr. Aug. 1, 1955. Decl. Feb. 7, 1957. 23p. Contract AT-11-1-GEN-8. $0.30(OTS).

A study has been made of the distribution of Pu and the principle fission products between molten U and Mg. Distribution coefficients were determined and in most instances, found to be fairly constant over a range of concentrations, extraction times, and uranium–extractant metal ratios. Equilibrium appeared to be attained in less than five minutes. This would indicate that in spite of small distribution coefficients a process for continuous extraction of Pu and fission products from molten U is quite promising. (auth)

789 NAA-SR-1697
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] APRIL–JUNE 1956. A. G. Buyers, W. J. Gardner, E. E. Motta, and G. Brand. Dec. 15, 1956. Decl. Mar. 13, 1957. 34p. Contract AT-11-1-GEN-8. $0.40(OTS).

Progress is reported in slug-cropping, electrorefining of Th, and volatilization of fission products. (For preceding period see NAA-SR-1676.) (T.R.H.)

790 NAA-SR-Memo-1299
North American Aviation, Inc., Downey, Calif.
A PROGRAM FOR A PYROPROCESSING DEVELOPMENT EXPERIMENT, INCLUDING REMOTE REFABRICATION OF REACTOR FUEL ASSEMBLIES. David W. Bareis, Kenneth L. Mattern, James R. Foltz, and Daniel J. Janeves. Mar. 30, 1955. Decl. Mar. 2, 1957. 42p. Contract AT-11-1-GEN-8. $7.80(ph OTS); $3.30(mf OTS).

A Pyroprocessing Development Experimental Facility (PDE) proposal is presented. Such a facility would be designed to permit the pyrochemical purification and remote refabrication of metallic U and Th fuel materials, and would furnish the necessary engineering data to permit full-scale processing plant design, realistic cost analyses, and the selection of optimized schemes for processing any metallic fuel. (L.M.T.)

791 NYO-1136
Vitro Mfg. Co., Pittsburgh.
PROBLEMS OF LEACHING AND DIGESTION OF URANIFEROUS SLAGS. Progress Report No. 2 [for] June 1951. H. Fleck. July 20, 1951. Decl. Feb. 28, 1957. 9p. Contract AT(30-1)-1134. $1.80(ph OTS); $1.80(mf OTS).

The volume of H_2 evolved on dissolving screened non-roasted U-bearing scrap, slag, and liner in HCl were determined; the values were found to indicate the degree of removal of metallic particles by screening. Digestions of nonroasted plant-pulverized scrap with HNO_3, $NaNO_3$, or $Fe_2(SO_4)_3$ in HCl and with HNO_3 or $NaNO_3$ in H_2SO_4 were not so effective as digestion with $NaClO_3$ in HCl. Doubling the free acid content and addition of $FeSO_4$ improved the efficiency of the latter digestion. Removal of CaF_2 from crude $(NH_4)_2\ UO_4$ by solution of the uranate in acetic acid, HCl, or H_2SO_4 was limited by slow filtration rates. (W.L.H.)

792 NYO-1159
Vitro Mfg. Co., Pittsburgh.
PROBLEMS OF REFINING URANIFEROUS RESIDUES. Progress Report for May 1953. H. Fleck and J. E. Summer. June 15, 1953. Decl. Mar. 12, 1957. 26p. Contract [AT(30-1)-1241]. $4.80(ph OTS); $2.70(mf OTS).

The applicability and refinement of various procedures for the treatment and digestion of C-slags and other uraniferous residues for recovery of U values were studied. The recovery of U from BFC-6 was investigated, placing emphasis on procedures which will not chemically alter the Cu and Sn media. Methods of U recovery from various residues resulting from Na_2CO_3 digestion of miscellaneous U-bearing materials, to reduce the U_3O_8 content below 0.05%, are discussed. (W.L.H.)

793 NYO-5030
Mallinckrodt Chemical Works, St. Louis.
REMOVAL OF SULFATE FROM N. G. LIQUOR. May 1, 1943. Decl. Feb. 26, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

In processing U ores, it is found that the ~1% sulfate concentrates in the N.G. (no good) liquor from the extractors. A method for precipitating and removing sulfates from these UNH liquors using $BaCO_3$ is given. (T.R.H.)

794 NYO-5085
Mallinckrodt Chemical Works, St. Louis.
REPORT ON THE PROCESSING OF PITCHBLENDE. Stanley H. Anonsen. Feb. 26, 1945. Decl. Mar. 8, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

A detailed description of the processing of pitchblende ores used at the Port Hope, Canada, refinery is given including both the recovery and purification of U and Ra. The health hazards present in the processing are discussed, and recommendations are made which will insure adequate health protection. (L.T.W.)

795 NYO-5160
Mallinckrodt Chemical Works, St. Louis.
A REPORT ON THE RECOVERY OF URANIUM FROM RESIDUES CONTAINING LARGE AMOUNTS OF PHOSPHATE. Edward G. Robles, Jr. Jan. 1946. Decl. Mar. 8, 1957. 24p. $4.80(ph OTS); $2.70(mf OTS).

The caustic leach method was investigated for its applicability to residue recovery. It was found that with a single leach upwards of 65% of the phosphate and 85% of the Mo present in the high-phosphate cake can be removed. Difficulties encountered with emulsion formation in nitrate solution extraction of cake thus treated were solved. Flow sheets and experimental results are given. (L.M.T.)

796 NYO-5171
Mallinckrodt Chemical Works, St. Louis.
A STUDY OF THE SALTING IN OF BARIUM AND LEAD SULFATES BY CHEMICAL AAA [PITCHBLENDE] LIQUORS AND PURE URANYL NITRATE SOLUTIONS. T. C. Furnas and A. E. Ruehle. Feb. 28, 1946. Decl. Mar. 7, 1957. 33p. Contract W-14-108-eng-8. $6.30(ph OTS); $3.00(mf OTS).

In the precipitation of $BaSO_4$ from $UO_2(NO_3)_2$ solutions the

effect of supersaturation accounts for only 2 to 5% as much $BaSO_4$ retained in solution as does the salting-in effect of the $UO_2(NO_3)_2$. The use of salting agents added to facilitate the ether extraction of the $UO_2(NO_3)_2$ has marked effects upon the precipitation of both $BaSO_4$ and $PbSO_4$ from the raffinates. The use of $Fe(NO_3)_3$ alone as the added salting agent gave considerably better precipitation characteristics than did a 1:1 mixture of $Fe(NO_3)_3$ and $Ca(NO_3)_2$. The use of Fe alone permitted only 7.2% as much $BaSO_4$ to precipitate immediately and only 80.7% as much total precipitation over a 24 hr period as did the use of the 1:1 mixture of salting agents. (L.M.T.)

797 ORNL-231
Oak Ridge National Lab., Tenn.
SEMI-WORKS DEVELOPMENT OF THE RaLa PROCESS. W. K. Eister. Sept. 20, 1949. Decl. Mar. 7, 1957. 80p. Contract W-7405-eng-26. $13.80(ph OTS); $4.80(mf OTS).

Semiworks development and demonstration for separation of Ba^{140} from irradiated Clinton U slugs was carried out to scale up the process from 200 to 2000 curies per run. (auth)

798 ORNL-621
Oak Ridge National Lab., Tenn.
ORNL RaLa PROCESS PILOT PLANT DEVELOPMENT. I. R. Higgins. May 17, 1951. Decl. Mar. 8, 1957. 39p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

Pilot-plant studies of the RaLa Process were continued. Filtration for the precipitation steps, and ion exchange for Ba purification were evaluated to increase yield, reduce process time, and improve product purity. (auth)

799 ORNL-1472(Del.)
Oak Ridge National Lab., Tenn.
DISTILLATION OF PUREX WASTES. R. G. Mansfield. Apr. 24, 1953. Decl. with deletions Mar. 5, 1957. 15p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Experiments on distillation of Purex waste containing fission products showed that the activity in the distillate is due principally to volatilized Ru. This can be minimized by keeping the still-pot nitric acid concentration below 9M. (auth)

800 ORNL-1873
Oak Ridge National Lab., Tenn.
MONEX PROCESS: TERMINAL REPORT R. J McNamee and R. P. Wischow. Jan. 31, 1956. Decl. Mar. 21, 1957. 41p. Contract W-7405-eng-26. $0.40(OTS).

Chemical and engineering data were obtained for the feed digestion system and the extraction-scrub step of the Monex TBP solvent-extraction process for recovering Th and U from HNO_3-digested unclarified monazite sludge. (auth)

801 ORNL-1941(Del.)
Oak Ridge National Lab., Tenn.
ORNL METAL RECOVERY PLANT PROCESSING CLEMENTINE REACTOR FUEL ELEMENTS. Terminal Report. J. L. Matherne. Sept. 7, 1955. Decl. with deletions Feb. 28, 1957. 19p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The processing of Los Alamos Fast Breeder (Clementine) Reactor fuel elements to recover Pu and Am is described. Chemical and equipment flowsheets are included. (C.W.H.)

802 ORNL-1994(Del.)
Oak Ridge National Lab., Tenn.
LABORATORY DEVELOPMENT OF THE THOREX PROCESS. Progress Report [for] July 1, 1954 to November 30, 1955. R. P. Wischow and R. G. Mansfield. June 22, 1956. Decl. with deletions Mar. 1, 1957. 60p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

Changes made in the Thorex process flowsheet were a decrease in the extraction column acidity to decrease Th losses and the addition of a second Th solvent-extraction cycle to provide the increased decontamination required when Th irradiated to 2000 to 4000 g of U^{233}/ton is processed. Bonded slugs could not be dissolved by the Thorex flowsheet procedure. Various laboratory-scale studies on feed preparation, first-cycle variables, and radiation damage to the solvent are reported. (auth)

803 ORNL-2015
Oak Ridge National Lab., Tenn.
ORNL METAL RECOVERY PLANT: PROCESSING OF ORNL GRAPHITE REACTOR FUEL ELEMENTS DURING THE PERIOD JULY AND AUGUST 1955. R. E. Brooksbank, J. M. Chandler, and C. D. Hylton. Nov. 11, 1955. Decl. Mar. 2, 1957. 33p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

The recovery of U and Pu from ORNL Graphite Reactor slugs in the Metal Recovery Plant is described. Modifications in the basic Purex Chemical and equipment flowsheets are outlined. (W.L.H.)

804 ORNL-2265(Del.)
Oak Ridge National Lab., Tenn.
AN ANALYSIS OF POWER REACTOR FUEL REPROCESSING. F. L. Culler, Jr., R. E. Blanco, H. E. Goeller, and C. D. Watson. Mar. 27, 1957. Decl. with deletions May 2, 1957. 133p. Contract W-7405-eng-26. $0.70 (OTS).

This report presents an analysis of the projected economics and processing capacity requirements for a power reactor fuel reprocessing industry based on the recovery of fertile and fissionable materials from presently proposed power reactors within the confines of the continental United States for the next five to ten years. An analysis of the present general state of development of a technology required for such an industry is given. A summary of results of power reactor reprocessing chemical and engineering development at Oak Ridge National Laboratory from July 1955 through December 1956 is given. (auth)

805 UCRL-1032
California. Univ., Berkeley. Radiation Lab.
THE CHELATE PROCESS. VI. PROCESS FLOW INVOLVING o-DICHLOROBENZENE AS THE SOLVENT FOR TTA. M. W. Davis, Jr., T. E. Hicks, and T. Verneulen. Jan. 1951. Decl. Mar. 21, 1957. 30p. Contract W-7405-Eng-48. $4.80(ph OTS); $2.70(mf OTS).

Comparative studies of a series of halogenated solvents, as carriers for TTA (thenoyltrifluoroacetone) in the chelate process for Pu extraction, indicate that ortho-dichlorobenzene most nearly satisfies the requirements. A complete process design is presented for use with this solvent, and flow data and equipment capacities are given for dissolver solution and for U-free fission product solution as alternate feeds to the process. Vertical mixer-settlers are recommended as the contactors, although pulsed columns or packed columns are also believed to be suitable. The size of such units is estimated from the best available rate and equilibrium data, and the effects of several operating variables are considered. (auth)

806 Y-636(Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
ISOTOPE RESEARCH AND PRODUCTION DIVISION PROGRESS REPORT FOR APRIL 1, 1950 TO JUNE 30, 1950. July 1, 1950. Decl. with deletions Mar. 6, 1957. 93p. Contract W-7405-eng-26. $15.30(ph OTS); $5.40 (mf OTS).

807 Y-766(Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
MATERIALS CHEMISTRY DIVISION PROGRESS REPORT

[FOR] JANUARY 1, 1951 TO MARCH 31, 1951. K. O. Johnsson, ed. May 21, 1951. Decl. with deletions Mar. 6, 1957. 28p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Spectroscopy

808 GAT-171(Rev. 1)
Goodyear Atomic Corp., Portsmouth, Ohio.
AN INTERPOLATIVE METHOD OF ASSAYING UF_6 WITH THE ISOTOPIC MASS SPECTROMETER. R. L. Smith, W. R. Shields, and C. D. Tabor. Apr. 2, 1956. Decl. Feb. 18, 1957. 24p. Contract AT(33-2)-1. $0.35 (OTS).

In the conventional procedure of isotopic analysis with the mass spectrometer, an observed ratio of mol ratios is corrected for an instrument phenomenon called memory. Accurate analysis requires accurate knowledge of this correction factor. To demonstrate this requirement, a brief description of the single standard method is given. Observed variations in this phenomenon within instruments and differences between instruments led to an investigation of a method which would reduce or eliminate the effect of memory. The essential features of an interpolative method are described, test results reported, and the direction of further study indicated. Comparative data are shown demonstrating the greater precision attained with the interpolative method. Error expressions for the single standard and interpolative methods also are presented. (auth)

Transuranic Elements and Compounds

809 CN-261
Chicago. Univ. Metallurgical Lab.
CHEMISTRY OF 94. Report for Month Ending September 15, 1942. Decl. Apr. 4, 1957. 61p. [Contract W-7401-eng-37]. A-296; CN-250 Contained in this report. $10.80(ph OTS); $3.90(mf OTS).

810 LA-2050
Los Alamos Scientific Lab., N. Mex.
SOME EXPERIMENTS ON THE PRECIPITATION OF PLUTONIUM COMPOUNDS FROM TRIBUTYL PHOSPHATE SOLUTIONS. W. J. Maraman and L. J. Mullins. June 11, 1956. Decl. Apr. 2, 1957. 8p. Contract W-7405-eng-36. $0.25(OTS).

A survey of the feasibility of precipitating Pu^{4+} compounds from 35% TBP in Gulf BT solution has been made. In general, the precipitations were made using an anhydrous salt, a saturated aqueous solution, and a dilute aqueous solution. The anhydrous precipitants which showed promise for further study were Na_2O_2 and NaF. Promising aqueous solutions were H_2O_2, NaF, HF, and K_2SO_4. The oxalate, hydroxide, iodate, phosphate, sulfite, and sulfide systems either gave precipitates of poor physical characteristics or resulted in high Pu solubilities in the organic phase. The precipitation of Pu^{3+} in the same solvent was not studied because of the relatively rapid oxidation of Pu^{3+} to Pu^{4+} in TBP solution containing HNO_3. (auth)

Uranium and Uranium Compounds

811 A-2901
Madison Square Area, Manhattan District, New York.
PERFORMANCE OF CHIPPED MAGNESIUM VERSUS HAMMERMILLED MAGNESIUM IN THE BOMB REDUCTION OF TETRAFLUORIDE. [PART] I. June 2, 1945. Decl. Mar. 14, 1957. 35p. $6.30(ph OTS); $3.00(mf OTS).

The O. Hommel Company, of Carnegie, Penna., supplied Mallinckrodt Chemical Works with three types of Mg chips or powder (Type 84, 85B, and 87B) formed by machining Mg ingots. This material was tested and compared with standard New England Lime Mg in a series of reduction runs. A new type of powder (Type 93) developed by the O. Hommel Company during the course of the first experiment was also sent to Mallinckrodt Chemical Works for preliminary testing. As this was superior to the other types tested, a large batch was made up and sent to both Electro Metallurgical Company and Mallinckrodt Chemical Works in order to confirm the preliminary results obtained by Mallinckrodt Chemical Works alone. (auth)

812 ANL-LB-SL-1092
[Argonne National Lab., Lemont, Ill.]
LITERATURE SEARCH ON CHEMICAL REDUCTION OF URANIUM OXIDES. G. A. Bennett. Aug. 23, 1956. Decl. Mar. 13, 1957. 11p. Contract [W-31-109-eng-38]. $3.30(ph OTS); $2.40(mf OTS).

Methods for the chemical reduction of uranium oxides which are discussed include reduction with calcium, magnesium, and carbon. Details of each method are included. 13 references. (C.H.)

813 HW-34118
General Electric Co., Hanford Atomic Products Operation, Richland, Wash.
ANODIC ETCHING PROCEDURE FOR URANIUM: USEFUL LIFETIME OF BATHS WITHOUT REGENERATION. E. C. Pitzer, Y. B. Katayama, and W. C. Budke. Dec. 14, 1954. Decl. Apr. 4, 1957. 10p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Electrodeposited Ni is found to be more adherent to a roughened than to a smooth U surface. This surface is obtained by anodizing the U in an aqueous solution of H_3PO_4–HCl acids. Lifetimes of this and the HNO_3 acid stripping solutions were studied. H_3PO_4–HCl solution anodized 3 sq. ft. of U without regeneration and the HNO_3 was satisfactory for at least one sq. ft. of U per liter. (F.S.)

814 HW-39457
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
FLUREX PROCESS. Interim Report. W. N. Carson, Jr. and W. P. Van Meter. Oct. 11, 1955. Decl. Apr. 5, 1957. 10p. Contract W-31-109-eng-52. $0.25(OTS).

The Flurex process, in a single unit operation, converts aqueous $UO_2(NO_3)_2$ into two products, an alkali uranic fluoride and HNO_3. This is achieved in an electrolytic cell which is divided into three compartments by an anion permeable membrane and a cation permeable membrane. $UO_2(NO_3)_2$ is introduced into the center compartment which is bounded by the two membranes. UO_2^{2+} migrates electrolytically through the cation permeable membrane into the cathode chamber which contains alkali and F^-; here it is reduced at a Hg cathode and alkali uranic fluoride precipitates. The NO_3 migrates through the anion permeable membrane to form HF in the anode chamber. Alkali uranic fluorides have been produced with current efficiencies of 80 to 100%. These salts are anhydrous and are readily dried in air. The Na product has been reduced to U in a Ca–I bomb reaction with yields of greater than 90%. Conversion to UF_6 remains to be investigated. (auth)

815 K-248(Del.)
Carbide and Carbon Chemical Corp. K-25 [Plant], Oak Ridge, Tenn.

U-235 CONTENT OF NATURAL URANIUM. PART I. [REPORT OF] WORKS LABORATORY DEPARTMENT. PART II. [REPORT OF] PHYSICS RESEARCH DEPARTMENT. W. W. Boardman and A. B. Meservey. Sept. 15, 1948. Decl. with deletions Feb. 25, 1957. 81p. Contract [W-7405-eng-26]. $12.30(ph OTS); $4.50(mf OTS).

Mass spectrometric comparisons were made of the relative isotopic abundances in natural U with the relative abundances in synthetic standards of known isotopic content. Results are presented from two simultaneous investigations by independent groups. Values obtained were 0.7113 and 0.7117 for the weight % of U^{235} in natural U. (C.H.)

816 K-404

Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.

CONVERSION OF URANYL FLUORIDE TO URANIUM TETRAFLUORIDE WITH GASEOUS REACTANTS. H. A. Bernhardt, E. W. Powell, and J. G. Schaffner. June 1, 1949. Decl. Feb. 23, 1957. 10p. (KLO-102). $1.80(ph OTS); $1.80(mf OTS).

Various methods were studied for the conversion of uranyl fluoride to uranium tetrafluoride with gaseous reactants. Hydrogen, ammonia, carbon tetrachloride, and dichlorodifluoromethane were used in combination with hydrogen fluoride or alone for the conversion. Of the various methods studied, the one in which the uranyl fluoride was first reduced with hydrogen followed by hydrofluorination with hydrogen fluoride gave the best conversion with the minimum number of undesirable features. (auth)

817 K-411

Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.

CONVERSION OF VARIOUS URANYL PHOSPHATES TO URANYL FLUORIDE BY MEANS OF ANHYDROUS HYDROGEN FLUORIDE. H. A. Bernhardt and S. Bernstein. June 1, 1949. Decl. Feb. 23, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The reactions of UO_2HPO_4, $NaUO_2PO_4$ and $NH_4UO_2PO_4$ with anhydrous HF are described. In the temperature range 400 to 500°C UO_2HPO_4 and $NH_4UO_2PO_4$ were converted to UO_2F_2, whereas $NaUO_2PO_4$ formed $NaUO_2F$ upon direct hydrofluorination. By the use of preliminary drying, thermal decomposition, and hydrofluorination it was possible to convert $NH_4UO_2PO_4$ to UO_2F_2, without the formation of the corrosive by-product, NH_4HF_2. (auth)

818 KLI-3588

Carbide and Carbon Chemicals Co. [K-25 Plant], Oak Ridge, Tenn.

SOLID-LIQUID EQUILIBRIUM AND SINTERING TEMPERATURES OF URANIUM TETRAFLUORIDE–URANIUM DIOXIDE MIXTURES. E. J. Barber. Sept. 15, 1955. Decl. Feb. 23, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Using the available experimental results and a reasonable estimate of the undetermined parameter, the solid-liquid equilibrium of the system UF_4–UO_2 was calculated. The results indicate that sintering may be important at temperatures as low as 500°C. (auth)

819 KLX-1068(Del.)

Kellex Corp., New York.

WASTE METAL RECOVERY PROCESS ENGINEERING SUMMARY REPORT FOR JULY 18, 1949–MARCH 1, 1950. Research Project 15D-1 (103). Mar. 7, 1950. Decl. with deletions Feb. 26, 1957. 8p. For General Electric Co. [Hanford Works]. W-31-109-eng-52 [Subcontract] G-148. $3.30(ph OTS); $2.40(mf OTS).

A flowsheet for the feed-preparation portion of a process for recovery for U from Hanford metal wastes was developed. Major steps include precipitation of the U as soda salt, metathesis to remove phosphate ion, and dissolution of the soda salt in 68% HNO_3. Work on this project was stopped March 1, 1950 following decision to develop a recovery facility based on an alternate process. (G.Y.)

820 LA-1952

Los Alamos Scientific Lab., N. Mex.

THE ACTIVATION OF LOW-REACTIVITY URANIUM DIOXIDE PARTICLES. R. J. Bard, D. L. Bunker, R. C. Greenough, and E. H. Kalmus. Oct. 1955. Decl. Apr. 1, 1957. 39p. Contract [W-7405-eng-36]. $0.40(OTS).

Low-reactivity UO_2 particles were treated by both physical and chemical means in attempts to improve their reactivity toward anhydrous HF. Grinding of the particles resulted in no appreciable activation until the particle size was reduced to the order of size of the primary crystallites. Partial dissolution of the particles in nitric acid resulted in some activation. Oxidation of low-reactivity UO_2 particles to U_3O_8 followed by reduction of the U_3O_8 with hydrogen yielded high-reactivity UO_2 particles from all materials studied. A detailed study of the activation of low-reactivity UO_2 particles toward HF by successive oxidation and reduction was carried out. The oxidation step, carried out with a number of maximum temperatures, was studied with respect to the effects of variations in the rate of heating from room temperature to the maximum temperature, in the amount of annealing of the oxidation product (U_3O_8) at the maximum temperature, and in the rate of cooling of the oxidation product. The reduction step was examined for effects produced by variations in the conditions of reduction at one particular reduction temperature. The activation of the particular low-reactivity UO_2 studied took place during the oxidation step, and the reactivity of the final UO_2 was determined by the conditions to which the antecedent U_3O_8 had been subjected. The effect of repeated successive oxidation and reduction on the reactivity of the UO_2 source material toward HF was evaluated by hydrofluorination. The reactivity increased progressively out to twelve activation cycles, at which point the measurements were discontinued. Oxide samples from this progression were characterized by means of specific surface, crystallite size, and bulk density measurements, as well as by electron microscopy. The crystallite size of the UO_2 remained essentially constant throughout. Activation of the UO_2 toward HF by repeated successive oxidation and reduction resulted from separation of the crystallites in the mosaic and from the formation of pores. There were no indications that any factors other than a decrease in the physical consolidation of the material contributed to the activation. (auth)

821 MCW-16

Mallinckrodt Chemical Works, St. Louis.

A PRELIMINARY REPORT ON A URANIUM–MOLYBDENUM COMPOUND. A. E. Ruehle and R. P. Smith. Apr. 1, 1946. Decl. Feb. 26, 1957. 4p. Contract W-14-108-eng-8. $1.80(ph OTS); $1.80(mf OTS).

Preliminary results are reported on the analysis and synthesis of a MoO_3-UO_2 precipitate formed during standing in some digestion liquors preceding ether extraction. The solubility of the precipitate in HNO_3 and H_2SO_4 and a possible method for recovering the U content of the precipitate is given. (D.E.B.)

822 NYO-1334

Mallinckrodt Chemical Works, St. Louis.

PILOT PLANT STUDIES OF LINER HARDENING, SECTION 4 OF THE SERIES; "URANIUM METAL BY BOMB REDUCTION." L. G. Weber. Apr. 1952. Decl. Mar. 2, 1957. 28p. Contract W-14-108-eng-8. $0.35(OTS).

Considerable exploratory research has been completed on

the bomb reduction method used at Mallinckrodt Chemical Works for the production of U metal. Full-scale equipment was utilized in a new pilot plant facility. Initial work was intended to yield information about the mechanism of the normal production procedure. Primary interest was directed toward gaining a better understanding of the effects of the liner upon bomb yield. A group of compounds was tested for their hardening action on dolomite liners. Metallic foil barriers, to prevent intermixing of solids, and to block the moisture in the dolomite from the charge, were investigated. Water scavengers in the dolomite, besides the normal fine Mg, were tried. The success of the use of ground slag (Mg fluoride) from previous bomb reductions dictated immediate curtailment of all dolomite liner work. The implications of the use of slag as a liner material are so broad, that maximum effort has been directed to developing a complete process for its use in production. (auth)

823 NYO-3792
Metal Hydrides Inc. Chemical Research Lab., Beverly, Mass.
FIRST QUARTERLY REPORT ON THE REDUCTIONS OF URANIUM OXIDES. Robert W. Bragdon. Oct. 10, 1952. Decl. Mar. 28, 1957. 13p. Contract AT(30-1)-1350. $3.30(ph OTS); $2.40(mf OTS).

The preliminary results of a program designed to evaluate methods for preparing uranium hydride from the various oxides of uranium are presented. The apparatus has been designed and constructed for the purpose of carrying out batch reductions under a variety of conditions employing calcium hydride, calcium metal and magnesium metal as the reducing agents. Since the separation of the alkaline earth oxides, present as by-products in the furnace products, promises to be the most difficult problem encountered in this investigation, a major portion of the time during this past quarterly period has been directed toward the development of a suitable leach method. No investigation of the reduction methods has been possible to date because of the inability to obtain SF materials. A brief outline of the future work is included. (auth)

824 NYO-5031
[Mallinckrodt Chemical Works, St. Louis.]
43-41 BY-PRODUCT RECOVERY—RECOVERY OF URANIUM FROM REFINERY RESIDUES. Preliminary Report No. 1. July 9, 1943. Decl. Feb. 26, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

Two possible methods for obtaining U from carnotite sludge are outlined. The feasibility of using each method is briefly discussed. The sludge which was available for experimental work when dried contained, 46.7% U_3O_8, 21.5% U_2O_3, 10.7% SO_4, 7.84% Na, and 4.1% Cl. The recovery of V from the carnotite sludge as a by product is also suggested. (auth)

825 NYO-5057(Del.)
Mallinckrodt Chemical Works, St. Louis.
GRAPHITE LINER PROGRAM. Report No. 5. Nov. 6, 1944. Decl. with deletions Feb. 26, 1957. 38p. $7.80(ph OTS); $3.30(mf OTS).

The addition of a small percentage of Mg fines to the Ste. Genevieve lime surrounding the graphite UF_4-reduction crucible appreciably reduced the firing time and temperature. Use of a U metal plate instead of Mg to support the charge did not improve or change the appearance of dogs. An attempt to use a steel sleeve in the mold to overcome the impossibility of removing the dogs from the fragile graphite resulted in complete fusion of the steel. The temperature distribution in the graphite crucible and mold was determined; definite evidence of a large temperature drop at (~1000°F) through the graphite wall was found. Variations in billets produced in tapered and untapered molds, in crucibles with various number, size, and arrangement of plugs in the pouring plate, and with various grades of Mg are illustrated. The number of times a graphite crucible may be used is discussed. (G.Y.)

826 NYO-5222
Mallinckrodt Chemical Works, St. Louis.
A STUDY OF WATER IN BROWN OXIDE [UO_3] AND ITS INFLUENCE ON THE OXIDATION OF THAT MATERIAL. S. H. Huston. Mar. 11, 1948. Decl. Mar. 8, 1957. 22p. $4.80(ph OTS); $2.70(mf OTS).

A study was made to determine the water content of UO_3 at all stages of handling in the plant and during storage, the rate of water adsorption, sources of water, and its influence on oxidation of the material. (L.T.W.)

827 ORNL-1985
Oak Ridge National Lab., Tenn.
INTERIM EVALUATION OF THE FLUOROX PROCESS FOR UF_4 AND UF_6 MANUFACTURE. J. E. Moore. May 26, 1955. Decl. Mar. 20, 1957. 10p. Contract [W-7405-eng-26]. (CF-55-5-176). $0.25(OTS).

The Fluorox Process for the conversion of UNH to UF_4 and UF_6, utilizing moving-bed techniques, is outlined. The Process involves the following steps: denitration of UNH to UO_3; pelleting of UO_3 with recycled UO_2F_2; reduction-hydrofluorination to UF_4; and oxidation of UF_4 to UF_6 product and UO_2F_2 intermediate. Advantages of the Fluorox Process include substitution of HF for F_2, reduction in HF requirements, and reduction in plant-size and mechanical complexity. (C.W.H.)

828 TID-5278
[Kellex Corp., New York.]
GENERAL PROPERTIES OF PROCESS GAS. [1943]. Decl. Mar. 1, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Tabulated and graphical data are given on the properties of UF_6, including specific heats, enthalpy, vapor pressures, and densities. (C.H.)

829 WAPD-TM-44
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
RESUMÉ OF URANIUM OXIDE DATA. IX. J. Belle and L. J. Jones. Mar. 15, 1957. Decl. May 3, 1957. 111p. Contract AT-11-1-GEN-14. $3.00(OTS).

Studies of UO_2 were undertaken to determine the characteristics of this oxide when used as a reactor fuel element component. Included in the investigation were chemical and physical properties, powder preparation and metallurgical studies, radiation effects and fission products, and fabrication of UO_2-Zr-Zr alloy clad systems. In connection with these studies an analysis was made of H-Zr reactions. Supplements are included which review the UO_2 studies being conducted at BMI and MCW. (D.E.B.)

Waste Disposal

830 KLX-1723(Del.)
Vitro Corp. of America, New York.
ECONOMIC EVALUATION OF POTENTIAL METHODS FOR CONCENTRATING REACTOR WASTES FOR STORAGE. STUDY OF IDAHO CHEMICAL PROCESSING PLANT WASTES. Interim Report [for] March 15-May 30, 1954. Job 1095. June 15, 1954. Decl. with deletions Feb. 20, 1957. 91p. Contract AT(30-1)-850. $15.30(ph OTS); $5.40(mf OTS).

Proposed methods were studied for reducing and storing wastes from the recovery of enriched uranium fuels at Idaho Chemical Processing Plant. Cost comparisons were made with underground tank storage. The limitations of

estimating storage costs for these highly corrosive wastes are explained. (L.M.T.)

ENGINEERING

831 CF-54-6-31
Oak Ridge National Lab., Tenn.
CONTINUATION OF HRT FUEL SOLUTION EVAPORATOR STUDY. C. L. Segaser. June 4, 1954. Decl. Apr. 2, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A third possible arrangement for the HRT fuel solution evaporator is shown. For this system, the evaporation is presumed to occur in 2 inclined 4-in. schedule 40 stainless steel steam-jacketed pipes supported below the inner dump tanks. In addition to its primary functions, the arrangement should also tend to keep the fuel solution in the dump tanks agitated. General specifications for the evaporator are listed. (W.L.H.)

832 CF-56-6-165
Oak Ridge National Lab., Tenn.
RESULTS OF PACKAGE BOILER TEST. HRT Test No. V 4a. J. L. Gory. June 28, 1956. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The package boiler was operated at a maximum steam rate of 1750 lb/hr at 1500 psi. The operation of the pumps, relief valves, gages, and pressure controller was checked. The time required for the boiler to come up to full operation was 15 to 20 min. On the basis of the test the boiler is considered to meet the manufacturer's specifications. (auth)

833 KAPL-1246
Knolls Atomic Power Lab., Schenectady, N. Y.
A RECENT MODEL OF THE MINIATURE MIXER-SETTLER. H. W. Alter and J. W. Codding. Dec. 15, 1954. Decl. Apr. 8, 1957. 17p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

A modification of the miniature mixer-settler is described. The bank is fabricated from stainless steel and fluorothene, and it can be used for elevated temperature operation. Changes in stage design increase phase separation and minimize back-mixing. Described also is a feed system, composed of Zenith Gear Pumps. Pictures and production drawings are included. (auth)

834 KAPL-M-WEJ-1
Knolls Atomic Power Lab., Schenectady, N. Y.
SOME MECHANISMS OF DRY BEARING CHATTER. W. E. Jahsman and D. R. Miller. June 4, 1956. 49p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

Much study has been devoted to the motion of shafts in lubricated journal bearings. In contrast, the work done on the motion of shafts in unlubricated or dry bearings is negligible. The reason is obvious: in most applications, lubrication of the bearings is relatively simple. This is not the case in nuclear power plant application where bearings operate in low viscosity primary coolants of liquid metal or water without the benefit of coolant contaminating lubricants. For this reason, an analytical study was made to determine what motions would be excited in a shaft mounted in a dry bearing. The study splits naturally into two parts: (a) the shaft maintains continuous contact with the bearing, either rolling or sliding or a combination of the two; (b) the shaft dances (or chatters) about the bearing in a series of impacts. Typical motions in each of these cases (rolling oscillations, sliding oscillations, stick-slip oscillations, chatter) are calculated and plotted. Except for the stick-slip oscillation, none of the motions studied is self-sustaining; rolling and sliding oscillations will damp out due to frictional and other losses, and chatter degenerates into a rolling whirl of the shaft around the bearing. (auth)

835 KLX-1038
Kellex Corp., New York.
REPORT OF THE INSTRUMENT AND MECHANICAL DEVELOPMENT DEPARTMENT ON REMOTELY OPERATED CONNECTORS. June 30, 1949. [For Hanford Works]. 98p. Decl. Mar. 22, 1957. Contract W-31-109-Eng-52, Subcontract G-148. $22.80(ph OTS); $7.20(mf OTS)

Development of pipe and electrical connector design appropriate to the unusual maintenance and operating conditions of the Redox Process is described. (K.S.)

836 KLX-1039
Kellex Corp., New York.
REPORT OF THE INSTRUMENT & MECHANICAL DEVELOPMENT DEPARTMENT ON LEAK-PROOF PACKLESS VALVES. Job 15. June 30, 1949. Decl. Apr. 19, 1957. 20p. [For Hanford Works]. Contract W-31-109-Eng-52, Subcontract G-148. $0.30(OTS).

Designs of bellows sealed valves that are leak-proof and fabricated of materials that will resist process solutions are presented. Detail drawings of 1 and ½-in. hand-operated valves, and 1 and ½-in. diaphragm-operated valves, have been prepared. The valve designs and the characteristics of certain critical components are described. The suitability and life expectancy of the component parts are being evaluated in a series of dynamic corrosion tests still in progress. (D.E.B.)

837 LRL-52
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
A FOUR-STAGE PROTOTYPE MIXER-SETTLER. J. L. Bloom and L. D. Christensen. Sept. 1953. Decl. Feb. 27, 1957. 48p. Contract AT(11-1)-74. $7.80(ph OTS); $3.30(mf OTS).

Information on the hydraulic stability, degree of agitation, phase separation qualities, and extraction efficiency as functions of physical changes in a four-stage prototype mixer-settler is presented. Suggestions are made for the design and construction of a mixer-settler for use in the Purex Process. (auth)

Heat Transfer and Fluid Flow

838 BMI-918(Rev.)
Battelle Memorial Inst., Columbus, Ohio.
A RECIRCULATING SUPERCRITICAL WATER LOOP. H. M. Epstein, W. H. Goldthwaite, J. W. Chastain, and S. L. Fawcett. Oct. 25, 1954. Decl. Mar. 12, 1957. 15p. Contract W-7405-eng-92. $3.30(ph OTS); $2.40(mf OTS).

A recirculating high-pressure stainless steel loop was developed for the study of heat transfer from small-bore tubes to supercritical water. It was found that under certain conditions, hot spots developed on the test section. The conditions leading to the formation of hot spots were investigated; the results of several runs are tabulated here. In developing the loop, several components were developed. These include a Dowtherm condensor, a steam accumulator, and a bellows-type pump. (auth)

839 CF-52-10-195
Oak Ridge National Lab., Tenn.
HEAT EXCHANGER DESIGN STUDY FOR THE ISHR. C. L. Segaser. Oct. 22, 1952. Decl. Feb. 15, 1957. 29p. $0.35(OTS).

The results of a study to determine reliable heat transfer data for the design of a suitable heat exchanger for the Intermediate Scale Homogeneous Reactor are presented. Design studies of a vertical submerged-tube evaporator with nucleate boiling from the outer tube surfaces and of a vertical tube, forced-circulation evaporator are presented in detail. (M.P.G.)

840 CF-53-10-208(Del.)
Oak Ridge National Lab., Tenn.
FLAT PLATE HEAT EXCHANGERS FOR REACTOR SYSTEM USE. R. W. Bussard. Oct. 26, 1953. Decl. with deletions Feb. 19, 1957. 29p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Equations are obtained relating the flow and heat transfer characteristics of counterflow flat plate heat exchangers, and design charts based on these equations are presented. The mechanical construction of such exchangers is discussed, with emphasis on the stresses induced by differential pressure loading of the plates and on the amounts and leak tightness of the required welds as compared to those in a typical tube and shell heat exchanger of comparable heat transfer capacity. Some general economic considerations and comparisons with tube and shell exchangers are also presented. (L.T.W.)

841 CF-56-4-204
Oak Ridge National Lab., Tenn.
CONVECTIONAL HEAT TRANSFER CONSIDERATIONS IN THE DESIGN OF FREEZE UNITS FOR HRT PROCESS LINES. S. H. Jury. April 10, 1956. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Equations are developed which define freezing times in Homogeneous Reactor Test fuel process lines with or without thermal convection. The equations are used to show thermal convection does not appreciably affect the freezing time for a $\frac{1}{4}$ in. × 6 in. freeze unit, but does tend to dominate freezing time in a 3 in. × 1 ft. unit. (D.E.B.)

842 KAPL-514
Knolls Atomic Power Lab., Schenectady, N. Y.
AN EXPERIMENTAL STUDY OF THE CONVECTIVE FLOW OF A HEAT GENERATING FLUID CONFINED IN A THIN VERTICAL CYLINDER. C. V. Moore. Apr. 14, 1951. Decl. Mar. 29, 1957. 15p. Contract W-31-109-eng-52. $0.25(OTS).

An experimental study of the geometry of convection flow inside a liquid-filled fuel pin is made by means of a tube filled with an electrolyte through which electric current is passed. It is found that at Reynolds' numbers of between 200 and 400 the flow changes gradually from a core type to a swirling type. With the swirling type of flow, the central temperatures in such a pin may be lower than would be predicted on the basis of core-type flow. (auth)

843 KAPL-M-DRM-1
General Electric Co. General Engineering and Consulting Lab., Schenectady, N. Y.
COMPARISON OF COOLANTS. D. R. Miller. Oct. 3, 1946. Decl. Feb. 14, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Liquid metal coolants Na, Na–K, Hg, Bi, Pb, and Pb–Bi are compared in terms of a figure of merit dependent only on cooling properties. (T.R.H.)

844 KAPL-M-LFE-4
Knolls Atomic Power Lab., Schenectady, N. Y.
CORROSION AND OTHER FACTORS IN THE APPLICATION OF SODIUM TO THE KAPL REACTOR. L. F. Epstein and C. E. Weber. Dec. 15, 1948. Decl. Mar. 25, 1957. 38p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf

Data accumulated on the nature of the engineering and technological problems involved in the use of liquid Na as a heat transfer medium in a nuclear reactor are presented. Some possible solutions are included. (M.H.R.)

845 KAPL-M-RGK-2
Knolls Atomic Power Lab., Schenectady, N. Y.
RESULTS OF SEVERAL WATER FLOW TESTS ON A FUEL ROD AND FAST BREEDER ROD. R. G. Kennison. Oct. 10, 1949. Decl. Feb. 27, 1957. 27p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

846 KAPL-M-RWL-5
Knolls Atomic Power Lab., Schenectady, N. Y.
RESULTS OF A SAPL THERMAL STRESS TEST ON A STAINLESS STEEL TUBE. R. W. Lockhart and R. G. Kennison. Dec. 15, 1949. Decl. Mar. 9, 1957. 7p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The effect of a repeating or cycling thermal stress on a thin wall stainless steel tube was determined. The apparatus used was essentially a paralleled flow single pass concentric tube heat exchanger with hot Na flowing through the inside of the inner tube and cold Na flowing in the annulus between the inner and outer tubes. It is concluded that the experimental values of Na film coefficients are about 35% lower than predicted but are of the right order of magnitude. A thermal stress varied from zero to 32,400 psi 132 times is insufficient to cause a crack or failure in a stainless steel tube at an average temperature of 540°F. (L.T.W.)

847 KAPL-M-SCT-10
Knolls Atomic Power Lab., Schenectady, N. Y.
PROJECT OWL: 8-INCH WATER INJECTION TEST. RUN NUMBER 1. Preliminary Results. C. W. Sorenson. Mar. 28, 1956. 7p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Sodium heated to 600°F and flowing initially at 2980 gpm was subjected to water injection at the rate of 0.024 lbs/sec. After about 124 sec the water injection was terminated because of greatly reduced Na flow. After ~10 min., a small leak was found in the loop at the point of water injection and the pressure in the loop was relieved by venting off gases. After the leak sealed itself, H_2 was removed by outgassing and the loop pressure raised to 20 psig by addition of N_2. When the loop was heated to 653°F a few hours later, flow returned to normal. Cold trapping was successfully initially at 500°F and eventually down to 300°F. A summary of flow, temperature, and pressure data are presented. (L.M.T.)

Materials Testing

848 HW-45649(Del.)
General Electric Co., Hanford Atomic Products Operation, Richland, Wash.
ULTRASONIC TESTING WITH LAMB WAVES. D. C. Worlton. Sept. 25, 1956. Decl. with deletions Mar. 5, 1957. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

A method is described whereby many types of flaws lying close to the surface of a metal can be ultrasonically detected, regardless of the time duration of the interrogating pulses. Lamb waves are established in the metal between a flaw and the surface by an ultrasonic beam which impinges at the proper angle of incidence. A suitably positioned receiver transducer picks up the waves to reveal the flaw. In this method the usually troublesome surface echo is eliminated from the receiver by an acoustic barrier, making it well suited for routine and automatic test-

ing. Results of applying the technique to several testing problems are discussed. (auth)

849 KAPL-M-EDL-71
Knolls Atomic Power Lab., Schenectady, N. Y.
SIR FREEZE SEAL TEST II. R. G. Jacoby and P. A. Benson. May 25, 1954. 31p. Contract [W-31-109-eng-52]. $7.80(ph OTS); $3.30(mf OTS).

Two arrangements of internal components of a freeze seal model were tested in order to determine the hydraulic characteristics. The internal configuration employing cylindrical knitted mesh packings in the connecting pipes proved the most suitable for the Na servicing freeze seal, because these packings are more easily fabricated. (auth)

850 KAPL-M-RMM-1
Knolls Atomic Power Lab., Schenectady, N. Y.
SHOCK AND VIBRATION DESIGN REQUIREMENTS FOR S3G/S4G. R. M. Mains. July 5, 1956. 13p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Shock and vibration tests on Submarine Advanced Reactor components (S3G/S4G) are reported. The design of a reactor vessel is controlled by shock requirements if its first-mode frequency is above 35 cps, and by vibration requirements if its frequency is below 35 cps. The design of auxiliary equipment mounted on a reactor vessel of 35 cps or greater first-mode frequency is controlled by shock requirements if the first-mode auxiliary frequency is above 35 cps, and by vibration requirements below 35 cps. The favorable range of frequency for the reactor vessel is 35 to 40 cps. A satisfactory shock load requirement for the preliminary design of auxiliary equipment is defined. The favorable range of frequency for auxiliary equipment is above 35 cps and 20% or more different from the reactor vessel frequency. Variations of 10% in first-mode frequency of the reactor vessel or the auxiliary equipment, or their frequency ratio, are significant since they cause disproportionately large changes in design requirements. (L.T.W.)

851 SEP-150
Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y. and Sylvania Electric Products Inc. Physics Research Labs., Bayside, N. Y.
NON-DESTRUCTIVE TESTING BY EDDY CURRENT METHODS. Final Engineering Report on Research. H. R. Holloway. Jan. 1954. Decl. Mar. 12, 1957. 158p. Contract AT(30-1)-gen-366. (YD-54-3). $0.70(OTS).

The Orthogonal Analyzer, an experimental eddy-current device for detecting flaws in the outer $\frac{1}{8}$ in. layer of fuel slugs, was developed and tested extensively. Non-destructive tests were made on 250 uncanned U pins from FMPC, giving indications of inhomogeneities on the surface and the subsurface. Portions of three FMPC pins were given destructive tests by compression at 125 to 200 tons, as a means of correlating, subsurface structure with eddy-current records from the Orthogonal Analyzer. Two gamma-extruded, warped slugs from Hanford were also inspected by the eddy current equipment and heavy striations were identified. Included is a review of commercially available eddy-current equipments for flaw detection and a summary of their relative merits. The design and operation of the Sylvania Orthogonal Analyzer is discussed in detail, and numerous eddy-current inspection records are shown. The high sensitivity and resolving power of this system is demonstrated and the possibilities of completely automatic slug inspection are pointed out. (auth)

852 WAPD-MP-75
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
DEVELOPMENTAL WORK ON NON-DESTRUCTIVE TEST METHODS OF INSPECTING SUBASSEMBLIES FOR CORROSION. Frank Cain, Jr. and Richard Alto. May 13, 1952. Decl. Apr. 15, 1957. 8p. $1.80(ph OTS); $1.80 (mf OTS).

Methods which could be used to distinguish between a corroded Zr surface and a non-corroded one are reviewed and evaluated. The major limitation was the proximity of the plates to be examined in the fuel subassembly. (T.R.H.)

Waste Disposal

853 KAPL-364(Pt. I)
Waste Processing Committee, AEC.
MEETING OF THE AEC WASTE PROCESSING COMMITTEE [HELD AT KNOLLS ATOMIC POWER LABORATORY], JUNE 26-28, 1950. PART I. J. A. Ayers, ed. Sept. 27, 1950. Decl. Feb. 25, 1957. 33p. Contract W-31-109-eng-52. $0.40(OTS).

The operation of facilities at KAPL for processing solid and liquid radioactive wastes are discussed in some detail with regard to both the evaporation and vacuum drying system for reducing liquid wastes to a minimum volume and the Hot incinerator for burning solid contaminated waste. The health physics aspects of waste processing are considered briefly. A crystallization recovery process has been developed for the recovery of $Al(NO_3)_3$ from Redox waste. The process is described, and the results are presented of a series of runs to determine its effectiveness. Results of an investigation of the crystallization separation of $NaNO_3$ from radiochemical wastes at ORNL are discussed. Possible applications of crystallization processes have been considered briefly. (F.S.)

MINERALOGY, METALLURGY, AND CERAMICS

854 BMI-267(Del.)
Battelle Memorial Inst., Columbus, Ohio.
URANIUM RECOVERY FROM A LOW-GRADE KATANGA SULFIDE CONCENTRATE. A. E. Bearse, R. A. Ewing, C. B. Sclar, and M. Pobereskin. Mar. 15, 1953. Decl. with deletions Feb. 13, 1957. 38p. Contract AT-30-1-Gen-228. $6.30(ph OTS); $3.00(mf OTS).

Methods for processing a low-grade sulfide concentrate from the Belgian Congo, containing approximately 10% U and approximately 70% Co, Fe, and Ni sulfides, have been investigated. Methods studied included: direct dissolution in HNO_3; the scrap plant process; leaching with dilute H_2SO_4; solvent leaching (TBP); and matte smelting. Maximum recoveries (over 99%) were obtained by aqueous leaching procedures. Uranium losses to the matte smelting amounted to only about 0.3 to 0.5%. Uranium recoveries on leaching the rather refractory matte-smelting slag reached a maximum of 98.5%, for an over-all recovery of 96.5 to 97.7% by this process. The results of a mineralogical examination of the concentrate are included. (auth)

855 KAPL-798(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
RESEARCH ON THE ELECTRODEPOSITION OF ZIRCONIUM, NIOBIUM, AND VANADIUM FROM NONAQUEOUS MEDIA. A. P. Beard, D. H. Ahmann, and F. J. Shipko. Aug. 22, 1952. Decl. with deletions Feb. 25, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The electrodeposition of Zr, Nb, and V from nonaqueous media was investigated as a possible method of coating U slugs. Research on solutions of salts of these metals in various anhydrous solvents is briefly described, but has been unsuccessful in providing an answer to the problems of electrodeposition of these metals from nonaqueous media. (C.H.)

856 KAPL-1139(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
A DEVICE TO REMOVE ALUMINUM JACKETS MECHANICALLY FROM BONDED URANIUM SLUGS. H. W. Alter, C. J. Beck, and B. V. Coplan. June 1, 1954. Decl. with deletions Feb. 25, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A description is given of a simple device to remove Al jackets from bonded U slugs. The successful application of the device to Hanford slugs is illustrated. (auth)

857 KAPL-M-DDC-1
Knolls Atomic Power Lab., Schenectady, N. Y.
NICKEL PLATING POROUS UO_2 STAINLESS STEEL DISPERSION TYPE FUEL CORES. D. C. Crooks. Oct. 26, 1956. Decl. Mar. 11, 1957. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A procedure for Ni plating on stainless steel–UO_2 dispersions which results in satisfactory clad bonding is given. (T.R.H.)

858 KAPL-M-DLD-1
Knolls Atomic Power Lab., Schenectady, N. Y.
APPLICATION OF CARBONYL PLATING TO THE COATING OF HANFORD FUEL ELEMENTS. D. L. Douglas. Feb. 11, 1952. Decl. Feb. 27, 1957. 9p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Of the several volatile metal carbonyl compounds only those of Ni and Mo show promise of being useful as plating materials for U fuel elements. In both cases it appears that a two-step process involving a coat of another metal will be necessary. Since the methods now being investigated appear to be more desirable, no experimental work on carbonyl plating is recommended at this time. (auth)

859 KAPL-M-HFB-1
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT ON THE FABRICATION OF THE FUEL CARTRIDGE. Herbert F. Brown and Robert C. Traxler. May 22, 1950. Decl. Feb. 26, 1957. 7p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

A metallurgical bond between Ti and U fuel rod elements by plating and brazing was successful. Ti was bonded to stainless steel by brazing and drawing in an inert atmosphere or in a vacuum at temperatures under 95°C to avoid diffusion. (F.S.)

860 KAPL-M-RCD-30
Knolls Atomic Power Lab., Schenectady, N. Y.
FLARING OF ZIRCONIUM END ADAPTERS. A. J. Ciancetta. [nd]. Decl. Feb. 28, 1957. 7p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

It is feasible to flare Zr end adapters to present designs for the control cylinder. A special flaring tool is used with two different size flaring dies in two operations. (F.S.)

Ceramics and Refractories

861 BMI-HRN-1
Battelle Memorial Inst., Columbus, Ohio.
QUARTERLY REPORT FOR JULY 1, TO SEPTEMBER 30, 1947. H. R. Nelson. Oct. 1, 1947. Decl. Mar. 18, 1957. 40p. Contract W-7405-eng-92. $7.80(ph OTS); $3.30(mf OTS).

The physical properties related to thermal stresses in ceramically fired BeO bodies were determined. The fabrication of BeO tubes by extrusion was studied. Portland cement concretes containing metallic Fe as aggregate were developed for reactor shielding. The properties of Al – U alloys were studied. The Al cladding of Th was also studied. Methods of bonding Ni to U were developed. (C.H.)

Corrosion

862 ANL-4763(Del.)
Argonne National Lab., Lemont, Ill.
SOME RADIATION CHEMISTRY ASPECTS OF THE CORROSION OF METALS ON IRRADIATION. W. L. Primak and L. H. Fuchs. Apr. 21, 1952. Decl. with deletions Mar. 14, 1957. 23p. Contract W-31-109-eng-38. $4.80 (ph OTS); $2.70(mf OTS).

Certain metals and alloys (e.g., Ni, Co, Pb, Al, inconel) which are subject to HNO_3 corrosion were found to form nitrates and basic nitrates when irradiated in moist air in reactors. Pb was found to form oxides when irradiated in moist oxygen or in water vapor. Some of the corrosion products appear as deposits in unexpected portions of the irradiation containers. The salts formed on the metals may lead to an unexpected spread of radioactivity. The corrosion may proceed to the point of producing serious damage to samples and parts in a nuclear reactor. The rate of nitrogen fixation was obtained in two reactors, CP-3 and X-10, and the yields calculated seem reasonable for the respective conditions of irradiation, contrary to some previous results reported. All the data concerning CP-3 refer to the former loading with natural U rods and to the thimble arrangement at that time. Some speculations on the genesis of some of the deposits found in irradiation cans are given. (auth)

863 CF-55-8-69(Del.)
Oak Ridge National Lab., Tenn.
EXAMINATION OF INCONEL-FLUORIDE PUMP LOOPS 4935-3 AND 4935-4. G. M. Adamson and R. S. Crouse. Aug. 12, 1955. Decl. with deletions Mar. 15, 1957. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The corrosion of the Inconel pump loops containing fused fluorides is evaluated. (L.T.W.)

864 CF-55-10-31(Del.)
Oak Ridge National Lab., Tenn.
EXAMINATION OF FLUORIDE PUMP LOOPS 4935-5 AND 4935-7. R. S. Crouse, J. H. DeVan, and E. A. Kovacevich. Oct. 5, 1955. Decl. with deletions Mar. 12, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The two gas-fired pump loops are a part of a series being run to determine the effect of wall temperature on mass transfer in an Inconel–fused fluoride system. The corrosion of the loops is evaluated. (L.T.W.)

865 CF-56-1-167
Oak Ridge National Lab., Tenn.
SOLUTION CORROSION GROUP QUARTERLY REPORT FOR THE PERIOD ENDING JANUARY 31, 1956. J. C. Griess, H. C. Savage, J. L. English, G. E. Moore, R. S. Greeley, R. S. Buxton, R. M. Warner, and R. M. Pierce. Jan. 31, 1956. Decl. Apr. 12, 1957. 51p. Contract [W-7405-eng-26]. $0.35(OTS).

A number of loop runs were made in order to determine how much H_2SO_4 is necessary to stabilize 0.04 m UO_2SO_4 solutions containing 0.005 m $CuSO_4$ at 300°C (HRT conditions) and to determine the corrosiveness of such solutions in the temperature range of 200 to 300°C. It was demonstrated that 0.01 m H_2SO_4 is not sufficient to prevent hydrolytic precipitation of Cu and U and that 0.02 m is suf-

ficient. Future tests may indicate that less than 0.02 m H_2SO_4 will be adequate. At 300°C corrosion of stainless steel is not serious, even with 0.02 m H_2SO_4 up to flow rates of 45 fps. However, at 200 to 250°C corrosion is more serious, and critical velocities as low as 15 to 20 fps were observed. Preliminary corrosion tests in a $UO_2(CO_3)_2$–$LiCO_3$ system showed that the corrosion rates of austenitic and ferritic stainless steels were low, in most cases lower than the rates observed in comparable UO_2SO_4 systems. In addition, carbon steel, which is completely dissolved by UO_2SO_4 solutions, showed low corrosion rates, 3 to 5 mpy. Further testing with highly concentrated UO_2SO_4 solutions containing added Li_2SO_4 confirmed previous results concerning the beneficial effects of the Li_2SO_4 on the corrosion of stainless steel. One loop run was with 3.8 m UO_2SO_4 and 3.8 m Li_2SO_4 at 350°C and showed the absence of a critical velocity up to 70 fps. Other runs made in toroids gave results similar to those obtained in loop runs. One series of toroid runs demonstrated that Ce_2SO_4 was at least as effective as Li_2SO_4 and perhaps even more effective. However, both loop and autoclave tests failed to demonstrate the abnormally low vapor pressure previously reported for such solutions. At present, there is no explanation for the erroneous results given in the last quarterly report. Static corrosion tests showed that type 1515N stainless steel has a corrosion resistance to UO_2SO_4 solutions about equal to that of type 347 stainless steel. On the other hand, an antigalling type alloy, designated as Waukesha Metal No. 88, was badly attacked (43 mpy) by UO_2SO_4 solutions at 300°C. In boiling 5% nitric acid, corrosion was also severe (32 mpy). Studies with Iso-Elastic spring alloy which had been Au-plated were carried out. At imperfections in the gold plate, attack was severe enough to allow the gold to peel off. Preliminary tests with another spring alloy, Elgiloy, showed it to be much more resistant than Iso-Elastic and indicated that it may be superior to Iso-Elastic for use in the pressurizer liquid-level controller. (auth)

866 DP-6(Del.)

Du Pont de Nemours (E. I.) and Co. Atomic Energy Div., Wilmington, Del.

ALUMINUM CORROSION–HIGH-PURITY WATER. N. A. Nielsen. May 1952. Decl. with deletions Feb. 15, 1957. 7p. Contract AT(07-2)-1. $1.80(ph OTS); $1.80(mf OTS).

A section of a 63S Al fuel tube with 2S and 43S and Type 304 stainless steel components exposed to water of high specific resistivity for 4 weeks did not show any corrosion pitting or abnormal corrosion. The test was carried out under conditions designed to simulate plant geometry and service conditions, such as water purity, temperature, and flow rate. (auth)

867 HW-17784

Hanford Works, Richland, Wash.

CORROSION OF STELLITE IN REDOX STREAMS. W. W. Koenig. May 15, 1950. Decl. Apr. 5, 1957. 18p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Depositions of Stellite Nos. 1, 6, 6-2A, 12, and 21 on austenitic stainless steel T-347 and depositions of Stellite No. 6 on T-309SCb were found to be resistant to Redox streams at room and boiling temperatures. They were not satisfactorily resistant to boiling 60% HNO_3. (auth)

868 HW-30386

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

INVESTIGATION OF REDOX D-12 WASTE CONCENTRATOR NO. 2 COIL NEST FAILURE. N. D. Groves and W. R. Smith. Mar. 2, 1954. Decl. Feb. 20, 1957. 20p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

869 HW-33075

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

INHIBITION OF FLUORIDE-CATALYZED CORROSION OF STAINLESS STEELS. R. L. Moore and R. A. Watts, Jr. Sept. 16, 1954. Decl. Apr. 4, 1957. 6p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

A series of experiments was conducted to demonstrate the magnitude of fluoride corrosive behavior and to test a number of substances as possible corrosive inhibitors. (F.S.)

870 HW-33641(Del.)

General Electric Company. [Hanford Atomic Products Operation, Richland, Wash.]

CARBON STEEL FOR HIGH TEMPERATURE REACTOR SYSTEMS. W. L. Pearl. Nov. 2, 1954. Decl. with deletions Mar. 14, 1957. 11p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

An evaluation of the feasibility of utilizing carbon steel as an alternate material for stainless steel in future reactor design is made. Problems and recommendations for future study are noted. (F.S.)

871 IDO-14195(Del.)

American Cyanamid Co. [Atomic Energy Div.], Idaho Falls, Idaho.

CORROSION OF ALUMINUM–URANIUM FUEL ASSEMBLIES IN STORAGE BASIN WATER. Investigation Order No. 103. W. J. Guay, T. L. Hoffman, and R. H. Perkins. Dec. 12, 1952. Decl. with deletions Feb. 6, 1957. 52p. Contract AT-(10-1)-177. $9.30(ph OTS); $3.60(mf OTS).

The corrosion of MTR and NP fuel assemblies stored in the SF building basin water of the Chemical Processing Plant was investigated. The principal causes of the corrosion were found by analysis of the basin water to be a chloride content of about 25 ppm and a relatively high electrical conductivity. Certain techniques that might reduce corrosion without a change in basin water composition were tested. (J.E.D.)

872 KAPL-952(Del.)

Knolls Atomic Power Lab., Schenectady, N. Y.

RESEARCH ON CORROSION RESISTANT COATINGS FOR URANIUM. Interim Report Covering the Period February 1952 to February 1953. C. E. Lacy, A. P. Beard, C. J. Beck, W. D. Davis, W. L. Robb, and G. E. Galonian. Decl. with deletions Feb. 26, 1957. 65p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

A general survey was made of possible techniques for applying a bonded, protective coating to uranium using a boiling water test to evaluate the coating quality. Of the techniques studied, the most promising methods for applying such a coating were electroplating with Ni, vapor plating with Zr, evaporation of Cr, and extrusion cladding with Zr. (auth)

873 KAPL-1450

Knolls Atomic Power Lab., Schenectady, N. Y.

REVIEW OF SIR PROJECT MODEL STEAM GENERATOR INTEGRITY. R. W. Lockhart, R. S. Kukfa, D. R. Miller, and W. L. Robb. Nov. 1, 1955. Decl. Mar. 13, 1957. 133p. Contract W-31-109-Eng-52. $3.25(OTS).

The operating conditions and post-operational examination of 26 new and rebuilt components that were arranged and operated as 11 SIR Project model liquid-metal-heated steam generators and heat exchangers are reviewed and correlated. Particular attention was directed toward operational damage to the units that either caused a leak or potentially would have caused a leak if the unit had been operated for a longer time. In addition to the operational data, a discussion of the state of the art of the corrosion and structural design technology and the necessity of hav-

ing an adequate structural design basis are given. All the leaks that were detected in the model tests were small in area, and when the fluids were mixed no excessive reaction was experienced; all fluids were contained except for leaks to the atmosphere. The third fluid leak-detecting systems were fairly successful in indicating a leak when the leak became of sufficient size to affect the detecting system. Additional development of these systems is in order to improve their sensitivity and reliability. To date, 28 potential leak areas plus 17 actual leak areas have been detected in these model heat exchangers. Recent models seem to be as prone to leak as earlier models. (auth)

874 KAPL-1570

Knolls Atomic Power Lab., Schenectady, N. Y.

CORROSION BEHAVIOR OF ZIRCONIUM-BASE URANIUM ALLOYS. U. Merten and D. C. Belouin. June 15, 1956. Decl. Mar. 28, 1957. 39p. Contract W-31-109-Eng-52. $0.40(OTS).

The corrosion behavior in high-temperature water of Zr-base alloys containing 7 to 8 wt.% of U was studied. A marked dependence of behavior on prior heat treatment was observed. In the annealed condition the alloys lost weight at rates of the order of one mg per dm^2 per hr of exposure to 360°C water. In the quenched condition the same alloys gained weight consistently, without any visible loss of oxide, for periods up to 4000 hr in length. The use of sponge Zr or a Zircaloy in place of crystal bar Zr in preparing the alloy had no noticeable effect on the corrosion resistance of the annealed material, but appeared to increase the corrosion rate observed after quenching. (auth)

875 KAPL-M-ELS-2

Knolls Atomic Power Lab., Schenectady, N. Y.

THE ANALYSIS OF KAPL-30 BY-PASS PURIFICATION RESIN COLUMN. E. L. Shirley, comp. Feb. 8, 1956. 11p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40 (mf OTS).

An analysis was made of the mixed bed resin column following removal from the in-pile carbon steel loop simulating operating conditions of a Submarine Advanced Reactor primary coolant loop. The residue deposited in the resin bed was mechanically separated from the resin, and the two fractions analyzed separately. Results are reported and procedures are discussed. (C.H.)

876 KAPL-M-UM-5

Knolls Atomic Power Lab., Schenectady, N. Y.

THE CORROSION BEHAVIOR OF SOME STAINLESS STEEL–BORON ALLOYS IN HIGH TEMPERATURE WATER. Donald C. Belouin and Ulrich Merten. Feb. 7, 1956. 15p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

A number of 18-24 stainless steels containing up to 2.3 wt. % B were exposed to static, high-purity water at 600 and 680°F. The results indicate only a small change in corrosion resistance under these conditions as a result of B additions. (auth)

877 WAPD-127(Pt. III)

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

DEVELOPMENT AND PROPERTIES OF URANIUM-BASE ALLOYS CORROSION RESISTANT IN HIGH TEMPERATURE WATER. PART III. CORROSION MECHANISM OF URANIUM-BASE ALLOYS IN HIGH TEMPERATURE WATER. M. W. Burkart, ed. July 1956. Decl. Aug. 7, 1957. 119p. Contract AT-11-1-GEN-14. $0.70(OTS).

The factors affecting corrosion resistance both of bare and of clad uranium–base alloys are reviewed and a mechanism proposed for their corrosion behavior. For unclad gamma-phase uranium alloys exposed to a high temperature water corrodent, it is proposed that the corrosion rate is determined primarily by the oxidation of the alloy by water. This behavior is contrary to that of alpha uranium in which the corrosion rate is primarily determined by the formation and subsequent oxidation of a nonadherent hydride layer. In gamma-phase alloys the hydrogen released by the corrosion reaction, rather than forming the thermodynamically stable UH_3 phase, dissolves (at least in part) in the base metal where it precipitates as a metastable hydride. The amount of hydrogen absorbed by the metal and hence precipitating as the metastable hydride may be markedly reduced by the addition of hydrogen depolarizers such as nickel or platinum to the water or to the metal. Similarly the amount of absorbed hydrogen may be reduced by introducing sinks that preferentially absorb hydrogen. Both alpha uranium and zirconium were shown to be suitable sinks. Precipitated hydride hardens and embrittles the matrix and by its preferential corrosion eventually leads to discontinuous failure. The hydride may be made to precipitate in a less harmful manner by heat treatments which precipitate nucleating and hardening impurities. The mode of the precipitation is shown to be sensitive to stress; in fact, the assumption of elastic stress as contributing to hydride precipitation is considered necessary to explain the distribution of the precipitate during corrosion. In order to apply a gamma-phase fuel alloy as a fuel element material, it is necessary primarily to reduce the amount of hydrogen absorbed by the fuel. This can be done by cladding the fuel with a zirconium-base alloy. Under such conditions it has been shown that fuel element lives in excess of 4 years should be obtainable with properly fabricated fuel elements before they are subject to corrosion failure. Optimum fuel element corrosion life can be achieved by decreasing the general corrosion rate, increasing the hydrogen solubility, increasing the hydrogen diffusion rate, and maintaining a proper clad-fuel bond. For uranium-base alloys such as U_3Si that do not form a hydride during hot water corrosion, use of a Zircaloy clad is unnecessary. Therefore, the development of alternate cladding materials such as Al for corrosion resistant fuel elements is predicted upon the development of uranium-base alloys that do not corrode by a hydride mechanism. In addition to U_3Si both Zr–U alloys and the strained alpha uranium-base alloys do not appear to corrode by a hydride mechanism. (auth)

878 WAPD-C-110

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

CRUD MEMORANDUM NO. 13. D. M. Wroughton. Jan. 12, 1953. Decl. Feb. 25, 1957. 21p. $4.80(ph OTS); $2.70(mf OTS).

Loop tests in the cyclotron indicated much less crud deposition with a pH of 10 than was obtained at pH 7 with no gas additions. Cleaning tests have not yet been completely satisfactory. Oxalic acid is so corrosive that there is now some question that it could be used in the reactor vessel alone as has been proposed. Tests in autoclaves and a small loop indicate a negligible adverse corrosion effect on metal couples from rubidium carbonate solutions if hydrogen is present. The small loop (M-4), running with hydrogen additions, continues to give good results. The rate of solubles corrosion product formation is essentially zero. The loop (M-1) running without degassification or gas additions, has built up a hydrogen content of 25 cc/liter and the rate of corrosion is approaching that of the loop with hydrogen addition. (auth)

879 WAPD-C-122

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

CRUD MEMORANDUM NO. 16. (Contains information through February 23, 1953.) D[onald] M. Wroughton.

Mar. 2, 1953. Decl. Feb. 21, 1957. 16p. $3.30(ph OTS); $2.40(mf OTS).

Corrosion data on various reactor materials from various loop experiments are presented. (L.M.T.)

880 WAPD-C-125
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRUD MEMORANDUM NO. 17. (Contains information through March 9, 1953.) D[onald] M. Wroughton. Mar. 12, 1953. Decl. Feb. 25, 1957. 16p. $3.30(ph OTS); $2.40(mf OTS).

881 WAPD-C-135
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRUD MEMORANDUM NO. 19. (Contains information through April 6, 1953.) D[onald] M. Wroughton. Apr. 9, 1953. Decl. Feb. 21, 1957. 17p. $3.30(ph OTS); $2.40(mf OTS).

Corrosion data on WAPD samples exposed to water containing 50 and 200 cc H_2 per liter are reported. The behavior of the Zr heat transfer tubes with respect to scale formation and removal during operation is described. The indication is that scale, once formed, cannot be removed by mere elimination of oxygen additions to the system. The effect of hydrogen on descaling will be studied. The BuShips Crud Loop, previously fouled with silica from a gage glass, is being cleaned. The WAPD loop data indicate that samples exposed to high pH water undergo approximately the same weight change as those exposed to hydrogenated neutral water, but the solubles corrosion rate of the loop systems appears to be much higher in the high pH loop. Preliminary quantitative information is presented on metal solubles having a bearing on system radioactivity. The procedure for analyzing oxygen in the presence of excess hydrogen has been worked out. Final examination of Battelle's static tests on iron solubility indicates that there is no significant difference in the solubility and corrosion of iron wire in water containing different gases. The early part of the program had indicated that solubility was greater in the presence of oxygen. Argonne's examination of in-pile corrosion samples indicates that the primary source of the gamma activity is Co^{60}. (auth)

882 WAPD-C-144
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRUD MEMORANDUM NUMBER 22. (Contains information through May 18, 1953). D. M. Wroughton. June 2, 1953. Decl. Mar. 18, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

883 WAPD-C-152
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRUD MEMORANDUM NUMBER 24. (Contains information through July 1, 1953). D. M. Wroughton. Aug. 17, 1953. Decl. Mar. 18, 1957. 28p. $4.80(ph OTS); $2.70(mf OTS).

884 WAPD-CP-489
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
HIGH NITROGEN AND OXYGEN IN BETTIS TEST FACILITY, MAY 25, 1954. Incident Report. Herman K. Lembersky. June 7, 1954. Decl. Mar. 18, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

The excessive O_2 and N_2 concentrations in Bettis Test Facility system H_2O is believed to be due to the introduction of 0.26 cubic feet of 1500 psi air during backflush of a hydraulic service system filter and entrapment of 0.29 cubic feet of air in the form of gas bubbles which was dissolving into the system-H_2O at an average rate of 2.6 cc/kg/hr. No definite conclusion can be made as to the origin or reason for existence of such gas bubbles. (W.L.H.)

885 WAPD-MM-448
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
REPORT OF THE MAY 1954 MEETING OF THE ZIRCONIUM ALLOY CORROSION COMMITTEE. S. Kass. June 1, 1954. Decl. Feb. 21, 1957. 115p. Contract AT-11-1-GEN-14. $21.30(ph OTS); $6.90(mf OTS).

The corrosion resistance of Zircaloy was investigated. The following conclusions were reached: the Sn content of Zircaloy-2 may be lowered with no effect on corrosion resistance; corrosion rate increases with increasing pressure, increasing N_2 content, and annealing temperatures over 900°C. The iron content may vary from 0.12 to 0.35% with little or no effect, and Ni is as effective as Fe in increasing the corrosion resistance of Zr and Zr–Sn. Cold reductions up to 50% improve the corrosion resistance of Zr. (T.R.H.)

Geology and Mineralogy

886 MITG-246
Massachusetts Inst. of Tech., Watertown, Mass. Mineral Engineering Lab.
LYOMETALLURGICAL TREATMENT OF SULFATED CARNOTITES AND OTHER ORES. Paul Galvanek, Jr. Feb. 21, 1951. Decl. Feb. 25, 1957. 40p. Contract AT-30-1-GEN-211. $0.40(OTS).

A new lyometallurgical method for extracting uranium from ores is described. The method involves static sulfuric acid-digestion at high solid-to-liquid ratios, treatment with calcium salts followed by leaching of the damp slurry with plain or nitric acid-containing acetone-ether or alcohol-ether mixtures. Very little nitrate is left in the residue. Nitric acid requirements are reduced through the use of calcium salts, in some instances being eliminated entirely. Uranium extractions ranged from 84 to 98% on Marysvale, cupriferous-uranium, and carnotite ores. Alumina may be used to remove impurities from pregnant solvents prior to uranium precipitation. Products assaying over 99.8% U_3O_8 have been obtained after alumina treatment. (auth)

Metals and Metallurgy

887 ANL-4399(Del.)
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT FOR OCTOBER, NOVEMBER AND DECEMBER 1949. Dec. 31, 1949. Decl. with deletions Feb. 12, 1957. 99p. Contract W-31-109-eng-38. $16.80(ph OTS); $5.70(mf OTS).

888 ANL-5043
Argonne National Lab., Lemont, Ill.
THE MANUFACTURE OF ALUMINUM CLAD ALUMINUM–PLUTONIUM ALLOY IRRADIATION SPECIMENS. Arthur B. Shuck. Apr. 17, 1952. Decl. Apr. 2, 1957. 33p. Contract W-31-109-eng-38. $0.40(OTS).

The manufacture of 25 aluminum-clad, 10 wt.% Al–Pu alloy and 5 wt.% Al–Pu alloy cored specimens is described. Steps in the manufacturing procedure include preparation of cores and cladding material, assembly and welding of compacts, out-gassing, roll bonding, shearing, identification marking, and final fabrication into napkin "rings." (auth)

889 ANL-RCV-SL-1180(Pts. I and II)
Argonne National Lab., Lemont, Ill.
METAL OXIDATION AND IGNITION. PARTS I AND II.

Feb. 12, 1957. Decl. Apr. 10, 1957. 62p. $7.80(ph OTS); $3.30(mf OTS).

Excerpts from Quarterly Progress reports of the Chemical Engineering Division of Argonne National Laboratory pertaining to spontaneous ignition of U, Zr, Th, and Pu are given. (M.H.R.)

890 BMI-1080(Del.)
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING MARCH 1956. Russell W. Dayton and Clyde R. Tipton, Jr. Apr. 2, 1956. Decl. with deletions Feb. 13, 1957. 87p. Contract W-7405-Eng-92. $12.30(ph OTS); $4.50(mf OTS).

Investigations were continued on the following: tensile properties of Al–U and Zr–U alloys at high temperatures; phase studies of U–Th alloys; mechanical properties of Zr and Zircaloy-2; bonding U slugs in Zr cans; radiation damage in graphite; extrusion cladding of reactor fuel plates; mechanical properties of dingot–U tubes; sinterability and electrical properties of UO_2; hydration of UO_3; corrosion of stainless steel by NH_3; heat treatment of U in fused-salt baths; development of refractory fuel elements; transformation kinetics, heat treatment, corrosion behavior, and radiation stability of U–Zr alloys; corrosion and heat treatment of Zr and Zr alloys; ceramic coating of U; physical and chemical properties of UN_x, US, and USi_x; quenching and aging of U–Ti alloys; oxidation resistance of Nb and Nb alloys; thermal stability of Zr hydrides; diffusion of O_2 in Zr alloys; corrosion of PWR fuel element cores; development of gas-cooled power reactor; and reflector control system for heterogeneous boiling reactors. (For preceding period, see BMI-1062.) (C.W.H.)

891 CF-50-6-185(Add. 1)
Oak Ridge National Lab., Tenn.
METALLURGICAL PROCESSES–ADDENDUM NO. 1. SF SUPPLEMENTAL MANUAL NO. 7. C. E. Larson. Nov. 7, 1951. Decl. Feb. 20, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

This manual describes the fabrication of MTR fuel elements up through the point beyond which there is no possibility of fissionable material loss except by loss of a fuel plate or fuel assembly. (auth)

892 CF-53-12-96
Oak Ridge National Lab., Tenn.
DEVELOPMENT OF BRAZE CLAD MATERIAL FOR CONSTRUCTION OF MTR TYPE FUEL ELEMENTS. J. E. Cunningham. Dec. 14, 1953. Decl. Mar. 15, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Reliable techniques were developed for fabricating a composite brazing sheet for use in construction of MTR type fuel elements. Brazing studies indicate the new product can be successfully used in the construction of the MTR fuel elements. (W.L.H.)

893 CF-54-5-74
Oak Ridge National Lab., Tenn.
DEVELOPMENT OF A METHOD FOR PREDICTING CONTINUOUS TRICKLE-TYPE COLUMN DISSOLVER BEHAVIOR FROM BATCH DATA. R. G. Wymer. May 14, 1954. Decl. Apr. 2, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A method for determining dissolver solution concentration as a function of length of column traversed for trickle-type continuous dissolvers has been developed. An illustrative example is worked out for the mercury-catalyzed nitric acid dissolution of U–Al cast alloy slugs. The method employs a graphical integration of simple functions of dissolution rates and acid concentrations. These rates and concentrations are easily obtained from batch experiments made over the important ranges of concentration. (auth)

894 CF-54-10-102(Del.)
Oak Ridge National Lab., Tenn.
RADIATION EFFECTS IN PRESSURE VESSEL MATERIALS AND WELDS. Paper Presented at Meeting of AEC Welding Committee on October 28, 1954 at Cincinnati, Ohio. J. C. Wilson. Nov. 15, 1954. Decl. Mar. 12, 1957. 23p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70 (mf OTS).

Radiation-induced changes in the physical and mechanical properties of reactor structural metals are discussed. Emphasis is placed on pressure vessel materials and welds. A theory is presented to explain some of the observed property changes. Data are included from studies on the effects of irradiation on Al alloys, carbon steel, Ni alloys, stainless steel, and carbon steel welds. (C.H.)

895 CF-55-4-163
Oak Ridge National Lab., Tenn.
MTR TEST NO. 3 OF APPR FUEL ELEMENT. R. J. Beaver and M. J. Feldman. Apr. 28, 1955. Decl. Feb. 22, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A method is discussed for sealing enriched plate sections by Cu brazing the sections in a stainless steel frame and sealing by heliarc welding prior to irradiation testing. (J.H.M.)

896 CF-56-4-56
Oak Ridge National Lab., Tenn.
IGNITION OF THORIUM METAL. E. S. Bomar. Apr. 10, 1956. Decl. Mar. 16, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The metal ignited in the course of preparing metal chips to be used as feed stock in a nonconsumable electrode arc melting furance. Preparatory to accumulating stock for the arc melting experiment, the Th slivers were pickled to remove surface oxide and stored in a glass storage vessel until needed. A time lapse of three weeks occurred before removal from the storage vessel. A technician began cutting the samples into thinner segments with a pair of hand shears when there was a flash of light and an accompanying explosion which blew the sample from his hands. A rapid decomposition of something associated with the surface of the sample was believed to have caused the accident. (W.L.H.)

897 CT-954
Ames Lab., Ames, Iowa.
TECHNOLOGICAL RESEARCH—METALLURGY—PART II OF REPORT FOR MONTH ENDING OCTOBER 2, 1943. Decl. Apr. 2, 1957. 42p. Contract [W-7405-eng-82]. (A-1378). $7.80(ph OTS); $3.30(mf OTS).

Progress is reported in the following studies: determination of the liquidus curve for U alloys having 0 to 4.8% C; the preparation and properties of the U–Cu, U–Sn, U–Bi, and U–Hg systems; the preparation of U by extraction of the hydride as amalgam; laboratory methods for the preparation of U and Th; the purification of Mg; determinations of B in lime; the effect of furnace temperature and firing time on the yield of U; a comparison of the properties of Mg from various sources; and methods for casting U metal. (C.H.)

898 CT-2116(Del.)
[Du Pont de Nemours (E. I.) & Co. Grasselli Chemicals Dept., Wilmington, Del.].
THE ELECTROPLATING OF TUBALLOY. A MANUAL OF OPERATING INSTRUCTIONS DEVELOPED FOR ELECTROPLATING ON TUBALLOY. Allen G. Gray. Jan. 31, 1945. Decl. with deletions Feb. 20, 1957. 65p. Contract W-7412-eng-24. $10.80(ph OTS); $3.90(mf OTS).

A manual of operating instructions for electroplating U

with any platable metal is presented including laboratory and commercial methods. Procedures are given for Ni, Cr, Fe, Cu, Zn, Cd, Pb, Sn, Ag, Au, Pd, and brass, and heat treatment to produce unusually impervious coatings is described. (T.R.H.)

899 HW-25608
[Hanford Works, Richland, Wash.]
METALLURGICAL EVALUATION OF THE URANIUM RODS ROLLED FOR THE CHALK RIVER PILE TEST. M. J. Sanderson and R. E. Hueschen. Sept. 11, 1952. Decl. Apr. 4, 1957. 15p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The metallurgical evaluation of the rolling of U rods at temperatures ranging from 552 to 660°C indicates that lower rolling temperatures increased the ultimate and yield strength, the amount of cold work, and the degree of preferred orientation existing in these rods. (F.S.)

900 ISC-144
Ames Lab., Ames, Iowa.
SEMI-ANNUAL PROGRESS REPORT IN METALLURGY FOR THE PERIOD OCTOBER 1, 1950–MARCH 31, 1951. H. A. Wilhelm. May 1, 1951. Decl. Mar. 13, 1957. 62p. Contract W-7405-eng-82. (deleted version: AECD-4046). $10.80(ph OTS); $3.90(mf OTS).

Data on the effect of dilution of tributyl phosphate with dibutyl ether, role of HNO_3, and salting-out effects of $Ca(NO_3)_2$ and $CaCl_2$ on the liquid-liquid extraction of Zr from Hf are listed. The effects of N_2 and O_2 contamination, quality of the ZrF_4 used, and relative quantities of Zn, Ca, and I_2 on the hardness and corrosion rates of Zr metal produced by bomb-reduction methods are discussed. A phase diagram, photomicrographs, and discussion of phase structure of the Sn–Zr alloy are presented. For the Zr–Nb alloy, a tentative phase diagram for up to 25% Nb is given, with photomicrographs and tabulations of results of hardness tests, tensile properties, and corrosion resistance to 315°C distilled H_2O of low-Nb alloys. Corrosion rates in 600°F water of Zr–Sb alloys containing 1 to 5% are listed. Preliminary results of measurements of the liquidus curve of the Zr–C system, a brief account of studies on the possible effects of impurities in experimental determination of the Zr–O system, and a tentative phase diagram of the Zr–Ge system are included. Modification of the Ames reduction process for Th metal by substitution of Zn metal and I_2 for $ZnCl_2$ is discussed. Corrosion of Th in 315°C water is described. Experiments in the electrolytic preparation of Ti and V powdered metal suitable for vacuum sintering are reported. (W.L.H.)

901 ISC-485(Del.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR JANUARY, FEBRUARY, AND MARCH 1954. Aug. 9, 1954. Decl. with deletions Feb. 25, 1957. 31p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported on the following investigations: separation of Nb and Ta, Ti and Nb, and Hf and Zr by solvent extraction; preparation of HfF_4, LaF_3, and vanadium chlorides; preparation of La, V, Nb, Hf, Th, and Ca; phase studies of U–Zn, U–Ag, U–Ag–Cr, Th–Zr, U–Nb, Zr–Nb, V–Zr, Zr–Zn, and Ta–Zr systems; corrosion of metals with liquid U–Cr alloys; crystal structure of $ZrGe_2$, sodium tungsten bronze, and Al–V systems; physical properties of metals from ultrasonic measurements; and development of refractories for melting vanadium. (C.W.H.)

902 ISC-601(Del.)
Ames Lab., Ames, Iowa.
COMPREHENSIVE REPORT ON PRODUCTION OF THORIUM. SECTION IA. THORIUM FLUORIDE PREPARATION. R. W. Fisher. July 1, 1953. Decl. with deletions Feb. 26, 1957. 12p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40(mf OTS).

The thorium production process includes the production of thorium tetrafluoride from commercial thorium nitrate, the production of thorium–zinc alloy from the thorium tetrafluoride, the production of massive thorium metal from the thorium–zinc alloy, and the processing of metal scrap. (C.H.)

903 KAPL-1617
Knolls Atomic Power Lab., Schenectady, N. Y.
THE DEVELOPMENT OF DISPERSION-TYPE REACTOR POISON COMPONENTS COMPOSED OF NEUTRON ABSORBING GLASS PARTICLES AS THE DISPERSED PHASE. G. L. Ploetz and W. E. Ray. Nov. 27, 1956. Decl. Apr. 10, 1957. 27p. Contract W-31-109-Eng-52. $0.35(OTS).

An investigation of CdB_2O_4 glasses to which high cross-section oxides (such as Gd_2O_3, Eu_2O_3, Sm_2O_3, Dy_2O_3, In_2O_3, and HFO_2) were added indicated that an extremely wide selection of vitreous reactor poison materials can be prepared. In this work, a glass containing 25 wt.% CdO, 25 wt.% Gd_2O_3, 30 wt.% B_2O_3 and 20 wt.% CaO was used to demonstrate the feasibility of fabricating metallic reactor components with this type of glass as the dispersed phase. Titanium was used as the matrix and cladding material in preparing model reactor components by the single hot-coextrusion technique. Satisfactory results were obtained. (auth)

904 KAPL-M-BMS-2
Knolls Atomic Power Lab., Schenectady, N. Y.
MAGNETIC PROPERTIES OF A 12% CHROMIUM MARTENSITIC STAINLESS STEEL. B. M. Smith. Feb. 21, 1956. 11p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The effect of various heat treatments on the mechanical and magnetic properties of type 403 stainless steel were investigated. Heat treatment varied from as received to 1200°C. In general, the higher temperature treatments resulted in higher strength and lower magnetic permeability. (auth)

905 KAPL-M-KOK-1
Knolls Atomic Power Lab., Schenectady, N. Y.
JOINING OF ZIRCALOY 2 TO AUSTENITIC STAINLESS STEEL. Report No. 1. K. H. Koopman and O. F. Kimball. July 31, 1956. 31p. Contract W-31-109-Eng-52. $4.80 (ph OTS); $2.70(mf OTS).

In order to utilize the advantages offered by Zircaloy 2 as a structural material in liquid cooled reactors a technique for producing a strong, ductile joint between Zircaloy and stainless steel must be developed. Welding, brazing, and cladding methods investigated in an attempt to solve this problem are described. (auth)

906 KAPL-M-KOK-3
Knolls Atomic Power Lab., Schenectady, N. Y.
SEAL WELDS. Status Report No. 2. K. H. Koopman. Aug. 20, 1956. 5p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Welds of good surface appearance were made with the sigma process using constant wire feed, small diameter 30 mil wire, and a constant potential rectifier-type power supply. The weld penetration was also uniform. Welds of uniform penetration were made at five times the speed of tungsten-arc welds. Tests indicated that the sigma process should be considered along with the tungsten-arc welding process for future seal welding applications, especially for those that can utilize the higher welding speed to advantage. (auth)

907 KAPL-M-RNH-3
Knolls Atomic Power Lab., Schenectady, N. Y.
A PROPOSED SYSTEM FOR THE MANUFACTURE OF HAFNIUM METAL POWDER BY THE HYDRIDE PROCESS. R. N. Honeyman and G. R. Kingsbury. Sept. 13, 1956. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A process is described for producing hafnium metal powder from hydrided Hf bar stock after pickling in HNO_3 to remove impurities. The brittle Hf hydride is ground to powder, dehydrided, and the resultant sinter cake broken up. A flow chart of the procedure is included. (C.H.)

908 KAPL-M-WGU-1
[Knolls Atomic Power Lab., Schenectady, N. Y.]
RECOVERY OF URANIUM CHIPS AND TURNINGS. W. G. Urbon. May 27, 1949. Decl. Feb. 27, 1957. 3p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Methods of preparing U scrap for shipment are considered. Bulk shipping of clean dry scrap is most desirable and conversion to the nitrate for liquid shipment is recommended where bulk shipment is not acceptable. (D.E.B.)

909 KAPL-M-WLF-4
Knolls Atomic Power Lab., Schenectady, N. Y.
EFFECTIVENESS OF VARIOUS TEMPERATURES FOR STRESS RELIEVING WELDED TYPE 347 STAINLESS STEEL. W. L. Fleischmann. Jan. 5, 1953. 15p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Welded-up ring samples of Type 347 stainless steel were given a number of different thermal treatments to study their effectiveness of relieving locked-up stresses. Temperatures of 1650 and 1850°F caused complete relaxation. At lower temperatures of 1100 and 1200°F the amount of relaxation obtainable over reasonable periods depended upon the magnitude of the residual stresses. Experiments conducted on twisted Type 347 wire showed that relaxation occurs at low temperatures. A treatment of 1100°F produces some untwisting and establishes a pattern of residual stresses which does not cause any further movement when the wire is cycled from room temperature to 850°F. (auth)

910 LA-1407(Del.)
Los Alamos Scientific Lab., N. Mex.
PREPARATION OF URANIUM–COLUMBIUM ALLOYS BY BOMB REDUCTION. B. R. Hayward. Feb. 1952. Decl. with deletions Mar. 7, 1957. 25p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

High purity, homogeneous alloys of U and Nb have been produced by adding powdered Nb metal or Nb_2O_5 to a regular reduction charge. Alloys up to 20.0 at. % Nb were produced. The preparation of the alloys, some of their properties, and casting results are described. (auth)

911 LA-1449(Del.)
Los Alamos Scientific Lab., N. Mex.
FUSION WELDING OF URANIUM. V. G. Nelson, J. M. Taub, and D. T. Doll. Aug. 1, 1952. Decl. with deletions July 24, 1957. 43p. Contract W-7405-Eng-36. $6.30(ph OTS); $3.30(mf OTS).

A technique for the fusion welding of uranium has been under development, and several methods of fusion welding have been investigated. The inert gas, shielded arc method has proven to be the most satisfactory of the processes thus far examined. Uranium properly welded by this method was free from porosity, cracks, and oxide inclusions. Certain precautions and some special techniques were required to make good welds. The tensile strength and yield strength of welded joints exceeded the strength of cast metal, but the ductility of welded joints was lower than that of cast metal. Uranium weldments exhibited distortion, but residual stresses appeared to be at a minimum due to the plasticity of the metal at temperatures of 250°C and above. All types of joints can be made in the flat position. Welding of uranium cannot be performed in the vertical or overhead positions. Uranium sheets in the thickness range 0.015 inch to 1 inch have been welded successfully. Welding must be performed with adequate ventilation of the welding area and with a satisfactory exhaust system available at the work if the health hazard is to be eliminated. (auth)

912 LA-1623
Los Alamos Scientific Lab., N. Mex.
SAFETY TESTS FOR MELTING AND CASTING [ORALLOY] [URANIUM ISOTOPES]. Frederick F. Hart. Dec. 1953. Decl. Apr. 1, 1957. 31p. Contract W-7405-eng-36. $4.80 (ph OTS); $2.70(mf OTS).

This report is a collection of data which were determined by several safety tests to assure safe procedures during the fabrication of Oy. These tests indicated that a 4 in. ID melting crucible which had become flooded from the cooling coils of the furnace would be critical with about 23.7 kg of Oy. A 5 in. ID crucible under similar conditions would be critical when it contained 21.3 kg. A series of measurements to simulate multiple crucibles in a melting furnace indicates that a flooded array of four charged crucibles with a spacing of 1.6 in. between the graphite heater jackets would have a neutron multiplication of 16.2. It has been shown that a 1 in. thick Oy plate in infinite paraffin tamping has a critical mass of 58 kg, and a $1\frac{1}{2}$-in. thick plate with the same tamping has a critical mass of 41 kg. Another test shows that a flooded Oy hemisphere 10 in. ID would be critical with a wall thickness of 1 in. at 60 kg. Also given in the last section are curves of Oy critical mass for various diameter cylinders with water, paraffin, and graphite tamping. (auth)

913 LAMS-626
Los Alamos Scientific Lab., N. Mex.
A METHOD FOR ROLLING URANIUM FOIL. George L. Barnard, Jr. Oct. 21, 1947. Decl. Mar. 18, 1957. 10p. $0.25(OTS).

Uranium-buttons may be rolled in a simple process by breaking and firing, plating, annealing at 600°C for $\frac{1}{2}$ hr in a low pressure vacuum furnace, rolling immediately while still hot and then annealing. If cross rolling is done, the piece can be given two passes 90° to each other at the same reduction without being annealed between passes. If a 10 to 20% reduction is wanted on every 90° turn, the piece should be annealed between each pass. Four passes at 5% with 200° heat treatment between each pass is not as good as one 20% reduction pass. (F.S.)

914 MIT-1052(Pt. I(Del.))
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
TECHNICAL PROGRESS REPORT FOR THE PERIOD APRIL THROUGH JUNE 1950. Sept. 8, 1950. Decl. with deletions Feb. 25, 1957. 74p. Contract W-7405-eng-175. $12.30(ph OTS); $4.50(mf OTS).

The latest production of flake Be does not exhibit as high ductility as was found previously. The oxygen content of Be powder of various particle sizes is determined to be proportional to the surface area per unit mass. Zr tubes (1-in. diam) with a 50-mil wall have been extruded. The lattice parameter of Zr has been determined from room temperature to 500°C. From this the various expansion coefficients have been deduced. Preliminary results on lattice parameter of Zr as a function of Hf content are given. The present status of Zr corrosion as a function of impurity content is described. A high-purity Zr–4% U alloy made by drip melting exhibited a weight change of 0.5 mg/cm^2 in 400 hr in 315°C water. The production and behavior of refractory

alloys as extrusion dies and tools are described. The construction and operation of apparatus for measuring internal friction of metals down to 4°K is described. (auth)

915 MIT-2018(Del.)
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
PROGRESS REPORT FOR JANUARY 1954. Jan. 27, 1954. Decl. with deletions Feb. 25, 1957. 5p. Contract AT(30-1)-981. $1.80(ph OTS); $1.80(mf OTS).

Progress of the work on Be, U, and Zr metallurgy, Al alloy development, and corrosion studies is briefly reported. (J.E.D.)

916 N-345
[Chicago. Univ. Metallurgical Lab.]
[METALLOGRAPHIC EXAMINATION OF ELECTROPLATED, Al–Si HOT DIPPED SLUG. F. Foote]. Dec. 10, 1943. Decl. Feb. 27, 1957. 5p. (MUC-FF-17). $1.80(ph OTS); $1.80(mf OTS).

917 NLCO-616
National Lead Co. of Ohio, Cincinnati.
PRODUCTION OF THORIUM METAL BY KROLL TECHNIQUE. T. C. Runion, A. H. Roberson, F. L. Cuthbert, and J. O. Davis. Mar. 6, 1956. Decl. Mar. 4, 1957. 28p. Contract AT(30-1)-1156. $0.35(OTS).

A report is presented on the preparation of thorium by the Kroll process. The work includes thorium chloride development studies carried out by the National Lead Company at Fernald, Ohio, and both chloride preparation and reduction studies at the Bureau of Mines, Albany, Oregon. (auth)

918 NMI-2045(Del.)
Nuclear Metals, Inc., Cambridge, Mass.
FUNDAMENTAL AND APPLIED RESEARCH AND DEVELOPMENT IN METALLURGY. Progress Report for April 1956. May 3, 1956. Decl. with deletions Feb. 27, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

A detailed report is presented on U metallurgy together with a brief reference to Nb-base alloys. (J.E.D.)

919 NYO-1133
Sylvania Electric Products, Inc., Bayside, N. Y.
ATOMIZATION METHOD OF MAKING URANIUM POWDER. Henry H. Hausner and Herman Mansfield. Aug. 7, 1950. Decl. Feb. 28, 1957. 16p. Contract AT-30-1-gen-367. $3.30(ph OTS); $2.40(mf OTS).

Atomized U powder was produced by forming an electric arc between two U electrodes in an inert atmosphere and sending a high velocity stream of inert gas through the arc. Uranium particles obtained by this method were of spherical shape; smaller particles contained mostly small grains, and larger particles were characterized by larger grains. The particles were ductile and could be hot-pressed to a compact of high density. The temporary equipment used for these preliminary tests on atomization was not adequate to control particle size. Suggestions for the production of atomized U powder of controllable quality are included. (auth)

920 NYO-3727
Horizons, Inc., Cleveland.
INVESTIGATIONS FOR THE PRODUCTION OF THORIUM METAL. Technical Progress Report for Third Quarter, November 1, 1952 to January 31, 1953. Merle E. Sibert. Mar. 9, 1953. Decl. Mar. 30, 1957. 31p. Contract AT(30-1)-1335. $4.80(ph OTS); $2.70(mf OTS).

921 NYO-3729
Horizons, Inc., Cleveland.
INVESTIGATION FOR THE PRODUCTION OF THORIUM METAL. Technical Progress Report — Fifth Quarter. M. Merlub-Sobel. Dec. 17, 1953. Decl. Mar. 28, 1957. 30p. Contract AT(30-1)-1335. $4.80(ph OTS); $2.70 (mf OTS).

Processes for preparing a suitable $ThCl_4$ electrolyte free of H_2O and O_2 were investigated including chlorination with $C-Cl_2$, sulfur chlorides, NaCl, and preparation from anhydrous solvents. The method selected consisted of reacting $ThOCO_3$ with HCl, adding NH_4Cl, and heating in inert atmosphere to drive off NH_4Cl. (T.R.H.)

922 ORNL-363(Rev.)
Oak Ridge National Lab., Tenn.
AIR CONTAMINATION AND ATTENDANT METAL LOSSES DURING PRODUCTION OF URANIUM–ALUMINUM ALLOYS. G. M. Adamson. July 12, 1949. Decl. Mar. 8, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

An investigation of the foundry air contamination and resulting uranium losses during the production and casting of U–Al alloys is reported. Tests were made using both enriched and natural U. Data show that although actual U losses are low, if they are not properly controlled, they can cause serious air contamination. A simple but adequate ventilating system is shown. (auth)

923 ORNL-1503(Del.)
Oak Ridge National Lab., Tenn.
METALLURGY DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1953. W. H. Bridges, ed. Apr. 6, 1953. Decl. Mar. 18, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

A study was made of the effect of the addition of V and Si on the properties of Th; corrosion of stainless steel in uranyl sulfate; corrosion, brazing, and welding of stainless steel; properties of Ti; method of combining deuterium and O for return to the reactor; spot welding of fuel elements; fabrication of reactor fuel elements; mechanical testing of U, Th, and stainless steel; surface rearrangements on a single crystal of Cu in contact with a saturated solution of Cu in Bi; development of cermets for reactor components; fabrication of Th–U fuel elements and Al–U alloy components. (J.E.D.)

924 ORNL-1778
Oak Ridge National Lab., Tenn.
PRELIMINARY INVESTIGATION OF THE FISSION PRODUCT RETENTION ABILITY OF CERMET COMPACTS. W. E. Moody, Jr., A. J. Taylor, and J. R. Johnson. July 8, 1955. Decl. Feb. 23, 1957. 22p. Contract W-7405-eng-26. $0.30(OTS).

A relatively simple comparative method for measuring fission product escape from solid fuel elements has been devised. Although the method has limitations, it serves to "screen" various materials as to their relative ability to retain volatile or gaseous fission products. The Xe, I and Te which were collected from the ceramic specimens employed in this study are probably representative of the fission products which are most likely to escape. On this premise it is shown that between 900 and 1000°C about 0.001% to 0.01% of the gaseous or volatile fission products are likely to escape from nonporous ceramics. In addition to the above fission products, others collected from tests made on loose UO_2 powder "standards" included Mo, Ba, Zr, Sr, and rare earths. (auth)

925 ORNL-1826(Del.)
Oak Ridge National Lab., Tenn.
DECONTAMINATION OF STAINLESS STEEL. D. O. Campbell. Mar. 2, 1955. Decl. with deletions Feb. 22, 1957. 45p. Contract W-7405-eng-26. $0.40(OTS).

The contamination on stainless steel exposed to Purex

process solutions is predominantly niobium, which is also the most difficult contaminant to remove. Zirconium accounts for a few percent of the contamination, and other fission products are relatively unimportant. Contamination of stainless steel with various dilutions of Purex waste followed the Freundlich adsorption isotherm relation between the amount of radioactivity adsorbed and its concentration in the contaminating solution. The addition of a fluoride solution of stable niobium and zirconium to the contaminating solution decreased the gamma activity adsorption by decreasing that of niobium and zirconium. Ruthenium then became relatively important. The most effective noncorrosive decontaminant was alkaline tartrate-peroxide. After the initial application of this reagent, its action was improved by a pretreatment of the contaminated surface with nitric acid. Several alternate treatments with nitric acid and alkaline tartrate-peroxide gave the best decontamination with negligible corrosion. The most effective decontaminant, 3% HF in 20% HNO_3, was severely corrosive to stainless steel. The rate of decontamination is reported for sodium hydroxide, alkaline tartrate, alkaline tartrate-peroxide, nitric acid, and oxalic acid. The reagents decontaminated rapidly at first, but the rate dropped to about 6% removal of activity per day after several days. High-intensity ultrasonic radiation increased the decontamination rate, but the problem of getting the radiation into large equipment would be formidable. A survey of the literature on decontamination is given. (auth)

926 ORNL-2029

Oak Ridge National Lab., Tenn.

DEVELOPMENT OF A CUBIC OXIDE PROTECTIVE FILM ON ZIRCONIUM. J. R. Johnson. Feb. 21, 1956. Decl. Mar. 2, 1957. 10p. Contract W-7405-eng-26. $0.25(OTS).

The cubic form of ZrO_2 has been formed by the air-oxidation of binary Zr alloys containing Nb, Y, or Ce in the temperature range 300 to 700°C. (C.W.H.)

927 SEP-56

Sylvania Electric Products Inc., Bayside, N. Y.

PREPARATION OF DISCRETE PARTICLES OF URANIUM NITRIDE AND CARBIDE FOR REACTOR ELEMENTS. Henry S. Rubenstein and Bernard Kopelman. May 4, 1951. Decl. Mar. 5, 1957. 25p. Contract AT-30-1-gen-366. $4.80(ph OTS); $2.70(mf OTS).

928 SEP-64

Sylvania Electric Products, Inc. Metallurgical Labs., Bayside, N. Y.

PRELIMINARY REPORT ON THE MANUFACTURE OF GADOLINIUM CONTAINING MATERIALS BY POWDER METALLURGY. Henry H. Hausner and Norman P. Pinto. Apr. 26, 1951. Decl. Mar. 5, 1957. 10p. Contract AT-30-1-GEN-366. $1.80(ph OTS); $1.80(mf OTS).

A brief account is given of the conversion of Gd_2O_3 into GdC_2, mixing of Gd compounds with Be, and the hot-pressing of the resulting mixtures. Micrographs show the structure of the hot-pressed compacts. (B.J.H.)

929 SEP-231

Sylvania Electric Products Inc., Atomic Energy Div., Bayside, N. Y.

INTEGRATED THORIUM–URANIUM TARGET ELEMENTS, Terminal Report. R. H. Witt, J. Nylin, and H. M. McCullough. Oct. 8, 1956. Decl. Mar. 14, 1957. 78p. Project No. 1878. Contract AT-30-1-GEN-366. $0.55 (OTS).

An investigation was made of the powder metallurgy of Th samples containing from 0 to 5% U. Densities approaching the calculated values for mixtures of the powders used were attained either by hot-pressing or by cold compacting, sintering, and coining. In addition, blending studies showed the advantages of using −325 mesh powders to obtain the best distribution of U particles in a Th matrix. Chemical analysis of various sections of samples indicated that uniform U contents can be obtained by control of the particle size and blending procedures. Cold compaction in vacuo or Ar showed little improvement in density over that obtained for samples cold compacted in air. Included is a comparison of the compactibility and sintering characteristics of Th powders produced by the hydride method and by C reduction of ThO_2. (auth)

930 SO-3004

Rensselaer Polytechnic Inst., Troy, N. Y. Powder Metallurgy Lab.

THE EXTRUSION OF COMPOSITE COMPACTS OF PARTICLES OF A HEAVY METAL IN AN ALUMINUM MATRIX. THE HOTPRESSING AND EXTRUSION OF ZIRCONIUM BY ELECTRICAL RESISTANCE HEATING. Final Report [Covering the Period July 1, 1950 to June 30, 1951]. A. B. Backensto, Jr., N. F. Hopson, and F. V. Lenel. June 30, 1951. Decl. Mar. 25, 1957. 92p. Contract AT(30-3)-37. $0.60(OTS).

Electrical resistance sintering was applied to the production of composite compacts consisting of particles of a heavy metal (Fe, Th, U) in an Al matrix and to the production of Zr powder compacts. Two procedures were used for producing the composite heavy metal–Al compacts. One consists of cold compacting a mixture of the powders, sintering the compact by electrical-resistance heating, and hot working the sintered compact. For the other procedure, the green composite compact was hot worked by extrusion without an intermediate-resistance sintering step. The highest percentage of the heavy metal possible in the composite compact which allowed extrusion of the compacts without defects was obtained by using a pure Al case around a core of the heavy metal–Al powder mixture. Resistance-sintered compacts of pure Zr powder having good ductility could be produced if the Zr powder was prepared, stored, and compacted without access to atmospheric gases. If these precautions were taken, the compacts were ductile regardless of whether the final resistance sintering was done in air or in an atmosphere of A. Discussions of the tensile properties of Zr compacts, the extrusion of Zr, the extrusion of green Mg powder compacts, and the resistance sintering of graphite are included in an appendix. (B.J.H.)

931 SRO-17

Horizons, Inc., Cleveland.

THE PREPARATION OF THORIUM METAL POWDER BY FUSED SALT ELECTROCHEMICAL TECHNIQUES. James L. Wyatt. [1955]. Decl. Mar. 15, 1957. 45p. Contract AT(30-1)-1335. (HZ-34). $7.80(ph OTS); $3.30(mf OTS).

A resume of development work done on the electrolysis of fused chlorides for Th production is given. (T.R.H.)

932 SRO-18

Horizons, Inc., Cleveland.

EXPANDED SCALE PROCESS DEVELOPMENT OF THE ELECTROLYTIC PRODUCTION OF THORIUM FROM $ThCl_4$ IN EXTERNALLY HEATED ELECTROLYTIC CELLS. Lothar Abraham, M. Merlub-Sobel, James L. Wyatt, and Eugene Wainer. Jan. 15, 1956. Decl. Mar. 14, 1957. 78p. Contract AT(30-1)-1335. (HZ-71). $12.30 (ph OTS); $4.50(mf OTS).

The commercial feasibility of the electrolytic process for Th production has been demonstrated. The cell produced 7 lb. of good quality metal per hour. A cathode cooling system was developed. The major construction materials, Ni and graphite, were satisfactory. The process yield was 86%. (T.R.H.)

933 TID-10016
Battelle Memorial Inst., Columbus, Ohio.
MECHANICAL PROPERTIES OF ZIRCALOY 2. G. T. Muehlenkamp and A. D. Schwope. Oct. 20, 1953. Decl. Mar. 18, 1957. 11p. Contract W-7405-eng-92. $0.25 (OTS).

Hydrogen contents, in amounts up to 400 ppm, had little effect on the tensile strength of Zircaloy-2; however, the tensile ductility was lowered considerably. The impact transition temperature increased with increasing amounts of hydrogen. The hot hardness and fatigue strength at 500°F were not affected by various amounts of hydrogen. (auth)

934 WAPD-FE-66
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
A REVIEW OF WELDING METHODS AND PROCESSES TO DETERMINE POSSIBLE END CLOSURE METHODS FOR PWR FUEL ELEMENTS. Experiment 11Q1. Earl L. Richards. Apr. 30, 1954. Decl. Apr. 15, 1957. 18p. $3.30(ph OTS); $2.40(mf OTS).

935 WAPD-FE-197
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
HARDNESS CONVERSION FOR URANIUM-BASE ALLOYS. Report PWR Experiments 6b2 and 6c2. W. L. Mudge, Jr. July 23, 1954. Decl. Mar. 19, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

Hardness conversion tables are presented for converting hardness values of various U-base alloys to a single hardness scale. (L.T.W.)

936 WAPD-FE-489
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
EXPERIMENT 10-m-2 EVALUATION OF THE BOND BETWEEN THE U–Mo FUEL AND CLADDING MATERIAL. H. R. Hoge. Oct. 11, 1954. Decl. Mar. 19, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

A compression method was devised for making bond tests and gave reproducible results. The bond between U–Mo core alloy and Zircaloy-2 is stronger than Zircaloy-2 but not as strong as U–Mo fuel alloy. The general effect of barrier layers in clad fuel rods is to decrease the bond strength. (L.M.T.)

937 WAPD-FE-681
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRELIMINARY PROCESS SPECIFICATION FOR MELTING URANIUM-12 WITHOUT MOLYBDENUM ALLOY. F. R. Lorenz. Mar. 1, 1955. Decl. Mar. 5, 1957. 5p. $1.80 (ph OTS); $1.80(mf OTS).

An external coil Micarta tube furnace using graphite crucibles for vacuum induction melting for U–Mo alloys is described and directions for operation are given. A remelt furnace is also described. (C.H.)

938 WAPD-FE-749
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].
BURSTING TESTS OF ZIRCALOY-2 TUBING. PART I. PRELIMINARY INVESTIGATION USING TUBES MACHINED FROM BAR STOCK. B. R. Price and L. F. Cochrun. Apr. 20, 1955. Decl. Mar. 11, 1957. 38p. $6.30(ph OTS); $3.00(mf OTS).

Data are summarized from burst tests on Zircaloy-2 tubes which are contemplated for use in PWR oxide fuel elements. Results show that the difference between the experimental strengths and calculated values can be explained on the basis of dimensional variations in the tube. The principal effect of annealing was found to be to reduce the yield and bursting strengths without necessarily influencing the resistance of the tube-to-crack propagation. (B.J.H.)

939 WAPD-FE-753
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRELIMINARY REFERENCE MANUFACTURING PROCESS FOR PWR FUEL ELEMENTS (U–Mo FUEL). Revision No. 2. J. G. Christ and F. R. Lorenz. Apr. 2, 1955. Decl. Mar. 5, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

An outline is given for the manufacturing process for (U–Mo) fuel elements including extrusion of components, Zircaloy-2 cladding, steel jackets, assembly of the element, rod stock extrusion and finishing, end caps and closings. (F.S.)

940 WAPD-FE-894
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
GASEOUS CONTAMINATION OF ZIRCALOY-2 CLADDING DURING FUEL ROD FABRICATION BY EXTRUSION AND DRAWING. H. J. Snyder. June 30, 1955. Decl. Mar. 14, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

Hydrogen, oxygen, and nitrogen analyses were performed on the Zircaloy-2 cladding of some fuel rods prior to hot extrusion, following hot extrusion, after cold drawing, and subsequent to nitric acid pickling. Analyses were also conducted on Zircaloy-2 which had not been subjected to extrusion or drawing, but only to pickling. Results of these analyses indicate that hydrogen, oxygen, and nitrogen are not picked up by Zircaloy-2 during the fabrication of fuel rods by the process presented. (auth)

941 WAPD-MDM-20
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
SUMMARY OF EXISTING KNOWLEDGE ON THE ZIRCONIUM–URANIUM ALLOYS (10 to 50 w/o URANIUM). W. L. Mudge, Jr. Dec. 8, 1954. Decl. Mar. 19, 1957. 20p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40 (mf OTS).

942 WAPD-NCE-2584
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
EXTRUSION OF STAINLESS STEEL CLAD URANIUM ALLOY. R. W. Tombaugh. Dec. 17, 1956. Decl. June 10, 1957. 16p. $3.30(ph OTS); $2.40(mf OTS).

The procedure and results of attempts to extrude stainless steel clad U–3.8 wt. % Si and stainless steel clad U–12 wt. % Mo are described. A theoretical discussion of the unsuccessful results is given. A correction value is used to adjust the "K" factor (resistance to deformation) to various extrusion ratios. (auth)

PHYSICS

943 BMI-1144
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING OCTOBER 1956. Russell W. Dayton and Clyde R. Tipton, Jr. Nov. 1, 1956. Decl. Mar. 15, 1957. 72p. Contract W-7405-eng-92. $0.50(OTS).

Research is reported in the development of materials for Hanford reactors, development, of Al-clad fuel elements, plant assistance to MCW, reactions between H_2O and Zr alloys, general fuel element development, Zr–U alloy studies, corrosion of Zr, reactor materials development, physical metallurgy, development of PWR, and Na–Ta compatibility at high temperatures. (T.R.H.)

944 KAPL-1302
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE NUCLEAR PHYSICS SECTION FOR DECEMBER 1954, JANUARY, FEBRUARY 1955. Decl. Mar. 30, 1957. 68p. Contract W-31-109-eng-52. $0.45 (OTS).

A new U^{235} fission chamber which is ready for installation is described. A 200-channel time-gated coincidence time analyzer has been developed. Mass spectrometry studies include brief discussions of a new spectrometer tube and better magnetic field control. Results are presented from studies on determining upper limits of "nonexistent" isotopes of the following elements: Li, Na, Mg, K, V, Mn, Nb, Ru, In, Cs, La, Pb, Ta, and U. Procedures for determining trace amounts of U and Pu in the atmosphere are briefly discussed. The contribution from the Compton effect in a γ spectrometer is canceled out by surrounding the spectrometer crystal with many NaI crystals which detect the Compton scattered quanta. The fuel assembly for the KAPL Plutonium Power Breeder is discussed. Most probable values for the parameters of the 3-ev resonance of Pu^{239} were derived. Radioactivity problems for open-cycle boiling water reactor systems were studied. (L.M.T.)

945 KAPL-M-RCD-20
Knolls Atomic Power Lab., Schenectady, N. Y.
A PREDICTION OF X-RAY ABSORPTION CHARACTERISTICS OF U–ZR ALLOY FUEL ELEMENT. W. R. Plant. Mar. 17, 1954. Decl. Feb. 26, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

946 ORNL-1926
Oak Ridge National Lab., Tenn.
PHYSICS DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 10, 1955. E. P. Blizard, ed. Sept. 7, 1955. Decl. Mar. 13, 1957. 23p. Contract W-7405-eng-26. $0.30(OTS).

The critical parameters of aqueous U^{235}-enriched solutions in cylindrical geometry were determined in several experiments. Data are presented for aluminum and stainless steel cylinders containing aqueous UO_2F_2 solutions. The cylinders were either unreflected, half-reflected, or surrounded on all sides by water at least 6 in. thick. Data are also presented on the effect of a latticed solid poison on critical mass and the critical parameters of U^{235}-enriched solutions in annular cylindrical vessels. The variation with concentration of the critical parameters of U solutions contained in spheres with concentration is being investigated. The critical height of an enriched UO_2F_2 solution contained in a cylinder was studied as a function of its distance from a concrete neutron reflector. Increasing the separation between the concrete and the cylinder resulted in an increase in the critical height. The fission cross section of natural U was measured from threshold to 4.0 Mev using protons to bombard a tritium gas target, and the data were corrected to obtain a cross section for U^{238}. U^{235} does not show a sharply defined threshold, but there is a measurable cross section at 500 kev which increases to 0.65 barn at 4.0 Mev. (T.R.H.)

947 TID-10062
Knolls Atomic Power Lab., Schenectady, N. Y.
THERMAL CONDUCTIVITY DATA FOR SOME NUCLEAR FUELS. L. R. McCreight. Oct. 17, 1952. Decl. Apr. 16, 1957. 19p. Contract W-31-109-Eng-52. $0.25(OTS).

Thermal conductivity data are reported for some mixtures of UO_2 with MgO, BeO, and Zr; U with Zr and Mg; several samples of B_4C; and two ceramic insulators. (auth)

948 UCRL-2053
California. Univ., Berkeley. Radiation Lab.
SPARK DAMAGE AND HIGH VOLTAGE BREAKDOWN OF METALS IN VACUUM AT 15 MEGACYCLES. Paul R. Byerly, Jr., Warren W. Chupp, and Harry G. Heard. Dec. 15, 1952. Decl. Mar. 5, 1957. 22p. Contract W-7405-eng-48. $4.80(ph OTS); $2.70(mf OTS).

Approximate breakdown voltages and spark damage estimates were made for Ag, Ti, Al alloy 52-SO, 316 stainless steel, and various combinations of Inconel and Cu. (L.T.W.)

Electrical Discharge

949 LA-2030
Los Alamos Scientific Lab., N. Mex.
STABILITY OF THE PINCH. Marshall Rosenbluth. Apr. 1956. Decl. Apr. 2, 1957. 36p. Contract W-7405-Eng-36. $0.40(OTS).

The stability of a pinched fluid is studied theoretically, under the combined influence of a longitudinal magnetic field and conducting shell. Conditions allowing complete stability are found. The method used to calculate the stability is to make the relevant virtual displacement and note if the resulting pressure distribution at the plasma surface is such as to cause the displacement to increase. (F.S.)

950 ORNL-2024
Oak Ridge National Lab., Tenn.
SOME EXPERIMENTS RELATING ION DIFFUSION IN A PLASMA TO THE NEUTRAL GAS DENSITY IN THE PRESENCE OF A MAGNETIC FIELD. Rodger V. Neidigh. June 15, 1956. Decl. Mar. 13, 1957. 13p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Gases

951 KAPL-M-MHS-21
Knolls Atomic Power Lab., Schenectady, N. Y.
CALCULATION OF THERMAL CONDUCTIVITY OF KRYPTON, XENON, AND FISSION GAS. M. H Shackelford. July 7, 1953. Decl. Feb. 25, 1957. 13p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The thermal conductivity of Kr, Xe, and fission gas mixture was calculated from the kinetic theory of gases. (L.T.W.)

Instruments

952 KLX-1041
Kellex Corp., New York.
REPORT OF THE INSTRUMENT AND MECHANICAL DEVELOPMENT DEPARTMENT ON LIQUID INTERFACE MEASUREMENT. Job 15. June 30, 1949. Decl. Mar. 25, 1957. 17p. [For Hanford Works]. Contract W-31-109-Eng-52, Subcontract G-148. $10.80(ph OTS); $3.90(mf OTS).

An interface-level detector for radioactive process systems, based on the principle of electrical induction, has been developed. The instrument possesses several advantages over pneumatic dip-leg equipment, and is adaptable to many other applications. (K.S.)

953 LA-1385
Los Alamos Scientific Lab., N. Mex.
A MODIFICATION OF THE NORELCO DIFFRACTOMETER FOR THE IDENTIFICATION OF PHASES IN METALLOGRAPHIC SPECIMENS. C. K. Stambaugh and D. D. Whyte. Feb. 25, 1952. Decl. Mar. 26, 1957. 9p. Contract W-7405-eng-36. $0.25(OTS).

A modification of the specimen support for the Norelco

diffractometer permits x-ray diffraction studies to be made of specimens prepared for metallographic examination. The unit has been successfully used to identify phases present in specimens of Pu-U alloys. (auth)

954 TID-10061
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
ELECTROMAGNETIC DETECTION OF Al–Si PENETRATION. H. L. Libby. Oct. 23, 1953. Decl. Mar. 18, 1957. 86p. Contract W-31-109-Eng-52. $0.50(OTS).

An electromagnetic device for detecting penetrations of the Al–Si braze layer into the aluminum envelope of the Hanford fuel element has been developed, and a prototype unit is being readied for production test. Operation of the equipment is essentially in agreement with the theory which has been developed in considerable detail. Laboratory tests indicate that the equipment is capable of detecting penetrations of Al–Si at least $\frac{1}{4} \times \frac{1}{4}$ inch cross section or larger that extend to within 0.020 inch of the fuel element surface. Smaller cross sections have been detected, but no measure of the minimum cross section which can reliably be detected has yet been made. Sensitivity is greater for penetrations nearer the surface; for example, penetrations at least $\frac{1}{16} \times \frac{1}{16}$ inch that extend to within 0.010 inch of the surface can be detected. The device uses a small sensing coil attached to a probe which scans the surface of the fuel element and which in conjunction with the electronic circuitry senses changes in the flow of eddy currents in the fuel element enclosure. Penetration of the envelope by the braze material has been shown to cause such changes. Scanning is accomplished by passing a fuel element rotating about its longitudinal axis under the sensing coil. Passage of a penetrated area under the sensing coil causes an alarm relay to operate. Laboratory tests were performed at speeds of approximately $\frac{1}{4}$ rpm. Preliminary tests have been made at 860 rpm, giving a complete scan of an eight-inch fuel element in about 10 seconds. (auth)

Isotopes

955 HW-28732(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
PRODUCTION OF Xe^{135} FOR LABORATORY PURPOSES. IRRADIATION OF URANIUM HYDROXIDE. Final Report on Production Test 305-15-P. J. W. Culvahouse, J. W. Finnigan, H. W. Lefevre, and R. E. Peterson. July 15, 1953. Decl. with deletions Feb. 19, 1957. 42p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

Isotope Separation

956 HW-39477
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
INTERIM REPORT ON THE SEPARATION OF ISOTOPES BY LIQUID THERMAL DIFFUSION. H. T. Hahn. Oct. 21, 1955. Decl. Feb. 26, 1957. 18p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Experiments in the separation of U isotopes by liquid thermal diffusion in $UO_2(NO_3)_2$–TBP complex and in aqueous and molten $UO_2(NO_3)_2$ are described. (T.R.H.)

957 Y-B20-78
Carbide and Carbon Chemicals Div. Y-12 Plant, Oak Ridge, Tenn.
ARC COMMITTEE MEETINGS FOR THE PERIOD JANUARY TO JUNE 1948. F. T. Howard, comp. Nov. 1950. Decl. Mar. 6, 1957. 215p. Contract W-7405-eng-26. $33.30(ph OTS); $9.60(mf OTS).

Minutes of meetings at which the various phases of calutron theory and operation were discussed are presented. Mathematical treatments of calutron electrical and magnetic characteristics and plasma and ionization behavior are included. (D.E.B.)

Mathematics

958 LA-731
[Los Alamos Scientific Lab., N. Mex.]
THE DETERMINATION OF EQUATION OF STATE DATA FROM MEASUREMENTS IN SHOCKS. J. E. Meyer. Oct. 1, 1948. Decl. March 5, 1956. 32p. Contract [W-7405-eng-36]. $6.30(ph OTS); $3.00(mf OTS).

Procedure is outlined for determining the constants in a reasonable empirical form of equation of state of solids from data on the isothermal compressibility at zero pressure, the coefficient of thermal expansion, and a single Hugoniot curve. The method is illustrated by application to Al. The exact equation relating the material velocity to the experimentally measured free surface velocity is worked out, since both material and shock velocities are needed to construct the Hugoniot curve. (auth)

Measuring Instruments and Techniques

959 LA-1487
Los Alamos Scientific Lab., N. Mex.
NEUTRON DISTRIBUTION MEASUREMENTS AT PAJARITO BY MEANS OF PHOTOGRAPHIC EMULSIONS. Dwight S. Young. May 15, 1952. Decl. Mar. 26, 1957. 37p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00 (mf OTS).

A method of using photographic emulsions for the study of neutron distributions is described. If a sheet of inert material capable of being made radioactive by neutrons is placed against a sheet of fissionable material and both are inserted in the neutron flux to be measured, fission fragments will be caught in the inert material. If then the inert material with fission fragments is placed against a photographic emulsion, the blackening at any point caused by beta activity from fission decay will be a function of the number of neutrons at that point. The three detectors which have given best results are U^{235}, U^{238}, and Au, the three being sensitive over a wide energy range. The construction of a simple machine for producing experimental critical assemblies in which the method has been used is reported. Sources of error are discussed. Curves of the perturbations when small spheres of various materials are placed in the center of the assembly are shown. (L.M.T.)

Neutrons

960 KAPL-M-RWD-11
Knolls Atomic Power Lab., Schenectady, N. Y.
METHOD OF AVERAGING THERMAL CROSS SECTIONS IN LIGHT WATER REACTORS. Robert W. Deutsch. Feb. 9, 1956. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A method of thermal neutron spectrum hardening which is based on a Wigner-Wilkins spectrum for a hydrogen gas is outlined. By use of this hardening scheme, the method of averaging thermal absorption and transport cross sections is shown. It was also found that the transport cross section of hydrogen in water has a 1/v dependence and a

cross section of 37.14 barns at a velocity of 2200 meters/sec. (auth)

961 LA-1272
Los Alamos Scientific Lab., N. Mex.
THREE-VELOCITY NEUTRON DIFFUSION CALCULATIONS FOR AN UNTAMPED ORALLOY SPHERE. Bengt Carlson. Oct. 13, 1951. Decl. Mar. 26, 1957. 15p. Contract W-7405-eng-36. $3.30(ph OTS); $2.40(mf OTS).

The results of a series of neutron-diffusion calculations relating to an untamped sphere are presented in detail. The three-velocity neutron-transport theory was taken as the basis for the analytical work preceding the computations. This theory, also known as the transport approximation, is defined in Report LA-1271. (D.E.B.)

962 UCRL-1730
California. Univ., Berkeley. Radiation Lab.
AGE OF Po–Be NEUTRONS IN UCRL WATER-LATTICE. John Ise. Mar. 18, 1952. Decl. Mar. 5, 1957. 6p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS)

Age theory was used to calculate the age of Po–Be neutrons in the UCRL water lattice, and calculated values were compared to experimental values. (W.L.H.)

Nuclear Physics

963 A-4716
Clinton Engineer Works, Oak Ridge, Tenn.
CRITICAL MASS STUDIES, PART I. Clifford K. Beck, Dixon Callihan, and Raymond L. Murray. June 10, 1947. Decl. Mar. 12, 1957. 44p. $0.35(OTS).

Using 1108 one-inch cubes of mock-up UF_6, containing 54 kg of U^{235} made from UF_4 (95% isotopic concentration of U^{235}) and $(CF_2)n$, and similar hydrogenous cubes of $(CH_2)n$, in various proportions and under various conditions of accumulation, 26 assemblies were built to criticality and four other assemblies were built which did not become critical. From these experiments, at least some information was obtained on the effect on critical mass of amount of intermixed hydrogen, uniformity of hydrogen-uranium mixing, density, geometry, shielding or reflector materials around the accumulation. Methods of preparation and properties of the materials used, description of experimental arrangements and procedures, and statement of data obtained, are given. A brief summary is given of the pertinent results obtained in the experiments. (auth)

964 UCRL-2387
California. Univ., Berkeley. Radiation Lab.
PILE NEUTRON PRODUCTION YIELD CURVES CALCULATED USING THE UCRL DIFFERENTIAL ANALYZER. R. J. Barrett, J. Killeen, J. O. Rasmussen, Jr., and S. G. Thompson. Oct. 30, 1953. Decl. Mar. 22, 1957. 108p. Contract W-7405-eng-48. $0.55(OTS).

A set of 69 pile yield curves for the production of Pu, Am, and Cm isotopes in a neutron flux of 5×10^{14} cm^{-2} sec^{-1} is given. The differential analyzer of the University of California Radiation Laboratory was used in obtaining the curves. The curves are given also on log-log plots for comparison. (auth)

Nuclear Properties

965 CF-56-2-106
Oak Ridge National Lab., Tenn.
RECENT FISSION PROCESS STUDIES WITH MONOENERGETIC NEUTRONS (TRIPARTITE CONFERENCE ON NUCLEAR CONSTANTS—JANUARY 10-12, 1956). J. L. Fowler. Feb. 21, 1956. Decl. Apr. 11, 1957. 17p. Contract [W-7405-eng-26].

The fission cross sections of U^{233}, U^{234}, U^{236}, and U^{238} were measured in terms of the U^{235} cross section from a few kev, or the effective threshold, to 4 Mev. Monoenergetic neutrons were from the V(p,n), Li(p,n), and T(p,n) reactions produced by protons accelerated by electrostatic generators. The published cross sections of U^{234} and U^{238} are compared with the cross section of U^{235}. The results for U^{233} differ somewhat from values reported in the literature. The absorption cross sections of U^{235} and Pu^{239} were measured at six energies in the 5 to 50 kev energy range. Neutrons from the reaction $V^{51}(p,n)Cr^{51}$ were used in spherical shell transmission measurements. The results show, in general, a decrease in absorption cross section with increasing energy up to about 30 kev and then a nearly constant value (up to 50 kev) of about 2.8 b for U^{235} and 2.2 b for Pu^{239}. The variation of ν for U^{235} with incident neutron energy was studied for the energy range 0 to 5 Mev. Neutrons were produced by the T(p,n) and the D(d,n) reactions. Fission neutrons were detected with a shielded recoil counter in coincidence with fission fragment pulses from an annular fission chamber placed around the recoil counter. The measurements were calibrated in terms of thermal ν by performing the experiment with essentially thermal neutrons. The results show an increase of ν of about 30% at 5 Mev. This is in fair agreement with the original prediction of this effect which was based on a phenomenological theory of evaporation of neutrons from the excited fission fragments. (auth)

966 KAPL-M-NCC-1
Knolls Atomic Power Lab., Schenectady, N. Y.
NUCLEAR CROSS SECTION STATUS REPORT. R. W. Deutsch, N. C. Francis, B. E. Simmons, and P. F. Zweifel. Apr. 16, 1956. 20p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

The current status of cross-section knowledge is discussed, and experiments and calculations for increasing this knowledge are described. A bibliography of available cross-section data is given, and recommendations for a program designed to provide more cross-section information are made. (auth)

967 NYOO-90
[New York Operations Office, AEC.]
DANGER COEFFICIENTS FOR IMPURITIES IN SEVERAL SUBSTANCES. E. Meservey. [1949?]. Decl. Mar. 19, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Sample calculations of danger coefficients and tabulated values for impurities in several materials are reported. (L.T.W.)

Nuclear Reactors

968 ANL-4439(Del.)
Argonne National Lab., Lemont, Ill.
MATERIALS TESTING REACTOR PROJECT. QUARTERLY REPORT FOR PERIOD ENDING MARCH 1, 1950. J. R. Huffman. Decl. with deletions Feb. 5, 1957. 44p. Contract W-31-109-eng-38. $9.30(ph OTS); $3.60(mf OTS).

Progress is reported in finalizing basic design data for the Materials Testing Reactor. The major emphasis at ANL was on issurance of design reports on practically all phases of the MTR project outside the reactor face and below the first floor level. Operation of the mock-up reactor at ORNL at 10 watts resulted in no major design changes. Topics discussed include the reactor building, wing, and reactor service building; canal and canal facilities; water systems; air exhaust systems; electrical power systems; effluent control; and shielding requirements. 11 drawings. (C.H.)

969 ANL-4847(Rev.)
Argonne National Lab., Lemont, Ill.
THE MANUFACTURE OF ENRICHED URANIUM FUEL SLUGS FOR THE EXPERIMENTAL BREEDER REACTOR. Arthur B. Shuck. Apr. 20, 1953. Decl. Mar. 18, 1957. 54p. Contract W-31-109-eng-38. $0.50(OTS).

The specifications, materials, and sequence of operations used to found and fabricate the first charge of enriched U fuel in the Experimental Breeder Reactor are described. Deviations from the standard process are discussed. Quantitative data are presented to show the over-all yield of fuel from the process and the quantities of material in the residues. The method of ignition of turnings is described together with quantitative data on the operation. (auth)

970 ANL-5012
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT FOR DECEMBER 1, 1952 THROUGH FEBRUARY 28, 1953. Mar. 15, 1953. Decl. Apr. 8, 1957. 34p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

Progress is reported on the following: design of a boiling reactor; the power breeder reactor (physics, thermal analysis, and experimental and development program); corrosion of Al coating; irradiation of wax and lubricants; effect of long-term irradiation on pile reactivity, caused by changing isotope ratios and fission products; Ni plating of Zr–U alloys; design and operation of a high-pressure, continuous-flow pH meter; stress-rupture tests on nuclear fuses for emergency release of nuclear poisons; and design of miniature fission pulse counters. (J.E.D.)

971 ANL-5060(Del.)
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] MARCH 1, 1953 THROUGH MAY 31, 1953. June 15, 1953. Decl. with deletions Jan. 17, 1957. 157p. Contract W-31-109-eng-38. $24.30(ph OTS); $7.50(mf OTS).

Power Breeder Reactor. Conceptual design studies for the Power Breeder Reactor indicate the feasibility of a contained primary coolant system in a package arrangement. One- and two-group calculations are presented in detail for preliminary PBR estimations. A technique for impregnating UO_2 with NaK has been developed. The resulting putty-type fuel was successfully extruded through a $\frac{3}{8}$-in. hole. Data are presented on the fuel element design factors imposed by heat transfer and cooling limitations for fuel plates, pins, and spheres. Fission heat liberated in the PBR blanket was estimated to be 7.7% of the total core power. Design criteria resulting from this estimate are analyzed. Heat exchanger size for the transfer of 500 Mw of heat from Na to NaK has been calculated for the intermediate exchanger and for the NaK-to-H_2O steam generator. A theoretical treatment is presented for the thermal stress analysis of stainless steel-clad U fuel plates. Boiling Reactor Heat Transfer Experiments. Further results have been obtained from natural-circulation boiling-density tests and investigations of natural-circulation flow rates. A special thermocouple design, meeting the rapid-response requirements of high-power plate transients, is described. Feasibility studies on heavy water boiling reactors are presented for spherical boiler, pressurized tube, and horizontal pressure tank configurations. Reactor Physics. Detailed theoretical methods are developed for determining long-time changes in reactors. Application is made to a one-group calculation of such changes in a spherically symmetric fast converter. Further data are reported from investigations of irradiation effects on the reactivity of U^{238}–U^{235} and Th^{232}–U^{235} systems. New data are available on the efficiency of the upper shield of the EBR, determined by neutron flux traverses taken on axes through the core center and inner blanket. Fuel Element Development. Attention has been directed to the study of U alloys with lower corrosion rates in water. Corrosion characteristics are given for UO_2, Zr-clad U plates, U, and Mg in diphenyl, and free energies for chemical reactions commonly occurring in reactor studies are tabulated. A study is reported on the pressure-drop characteristics of fuel elements with variations in flow rate, metal-to-water ratio, pitch, and water density. The increased resistance of such fuel wires was found to be less than anticipated. Radiation Studies. A comparison is made between the radiation effects of Dow-Corning high-vacuum silicone stopcock grease and Standard Oil Co. Lg-0147 grease. (C.H.)

972 ANL-5061
Argonne National Lab., Lemont, Ill.
FEASIBILITY REPORT ON FAST EXPONENTIAL EXPERIMENT. R. O. Brittan, H. H. Hummel, J. J. Livingood, F. H. Martens, and B. I. Spinrad. June 10, 1953. Decl. Feb. 12, 1957. 32p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

The general program established at Argonne National Laboratory in connection with the Fast Power Breeder Reactor (PBR) includes performance of exponential experiments on assemblies having compositions which may exist in the reactor core. This report deals with that phase of the program known as the Fast Exponential Experiment which may be described very briefly as follows: An assembly of fissile U^{235} and fertile material (too small to be self-critical) intermixed with poisons (such as are encountered in the mechanical structure and coolant system of a self-sustaining reactor) is fed with neutrons from an auxiliary source. By measurements of the neutron flux within the assembly, important parameters can be calculated which are necessary to the proper design of an actual critical reactor of the same composition. (auth)

973 ANL-5327(Del.)
Argonne National Lab., Lemont, Ill.
DESIGN STUDY OF SMALL BOILING REACTORS FOR POWER AND HEAT PRODUCTION. M. Treshow. Nov. 1954. Decl. with deletions Mar. 6, 1957. 63p. Contract W-31-109-eng-38. $0.40(OTS).

A design study has been made of a small "Package" nuclear power plant for the production of electric power and heat in remotely located, inaccessible areas devoid of natural fuels. The design utilizes a horizontal boiling reactor as a steam generator consistent with safe and simple equipment and a minimum building height. A reactor design of $5\frac{1}{2}$ Mw capacity, with a combined net electric power output of 750 kw and a heat plant output of 4500 kw, was studied in detail. Tentative cost estimates are presented on the basis of this combination. General comparisons have been made between different systems designed for either independent or combined production of 425 kw net electric power and 2500 kw available heat. (auth)

974 ANL-5345(Del.2)
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT FOR PERIOD JULY 1, 1954 THROUGH SEPTEMBER 30, 1954. Oct. 15, 1954. Decl. with deletions Mar. 29, 1957. 127p. Contract W-31-109-eng-38. $19.80(ph OTS); $6.30 (mf OTS).

Reactor Programs. The design of the Boiling Experimental Reactor is described. The Borax-I reactor was intentionally destroyed by suddenly adding 4% excess reactivity. Design of the Borax-II reactor is described. It

will be used for studies of transient behavior and steady-state operation as a steam producer. A preliminary design configuration has been established in EBR-II employing central, inner, and outer blankets. Fuel geometries including a smaller pin-type element are being investigated. Supporting Design, Research, and Development. The accuracy of diffusion theory for fast reactors was investigated. Water corrosion tests have been performed on high-U alloys, Zr- and Ti-base alloys, and Ni-plated Al plates. Thermal cycling tests were performed on fuel rods and pin-type fuel elements. Results are reported on tests on a current conductor connection for a high-temperature d-c electromagnetic pump, on Na valves with conventional packing, and on a rod-gripper test facility for the EBR-II. A vacuum cup sampler for high-temperature Na and NaK systems is described. Lifting and loading mechanisms for the EBR-II working model are described. (T.R.H.)

975 ANL-5398(Del.)
Argonne National Lab., Lemont, Ill.
REACTOR DEVELOPMENT QUARTERLY PROGRESS REPORT. Jan. 31, 1955. Decl. with deletions Jan. 21, 1957. 63p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

The design and construction program of the Boiling Experimental Reactor is reviewed. A number of preliminary experiments were performed with Borax-II at pressures between 75 and 300 psi. The most corrosion-resistant U–Zr–Nb alloy developed so far is produced by heating in a vacuum to the γ phase, quenching, and aging for 2 hr at 400°C. Special attention is given to the removal of H_2 from the material. Unclad and unirradiated samples of U–Nb and U–Nb–Zr alloys were corrosion tested in H_2O. Corrosion rates were also measured under irradiation conditions in CP-5. Elongation measurements of irradiated wrought and cast U–Zr material suggested no way for treating the wrought fuel so that stability comparable to the cast material could be obtained. Natural circulation boiling density tests at 600 psia were made in order to determine the effects of channel cross section and subcooling on the steam void fraction. Results of autoclave and dynamic corrosion studies of 2-S Al in H_2O are reported. These results include the testing of Ni-clad samples. A large number of criticality calculations were performed for the EBR-II and the PBR. The solubility of ThO_2 pellets containing various concentrations of U_3O_8 was tested in water at 316°C for periods of 672 to 744 hr. None of the samples disintegrated, although at least one sample developed cracks. Solutions of reactor kinetic equations were attempted for the purpose of studying transients in reactors with lifetimes of 7×10^{-9}, 10^{-7}, and 6×10^{-5} sec. Ignition experiments were performed on Th, Cu, Al, Fe, Mg, Zr, and fluorothene when contacted with fluorides. Except for Zr and fluorothene, the materials did not ignite. (C.H.)

976 ANL-5492(Del.)
Argonne National Lab., Lemont, Ill.
FAST REACTOR PHYSICS CALCULATIONS. Robert Avery. Jan. 1956. Decl. with deletions Jan. 17, 1957. 80p. Contract W-31-109-eng-38. $13.80(ph OTS); $4.80(mf OTS).

Multigroup diffusion theory calculations performed in connection with the fast power breeder reactor development program at Argonne are summarized. The problems investigated relate to: steady-state properties; long-term changes; reactivity coefficients associated with temperature changes; effectiveness of various control methods; evaluation of neutron lifetime; various methods of radial power flattening; various parametric studies to determine optimum design points; accuracy of diffusion theory for systems under consideration; and analysis of experimental results. (auth)

977 ANL-5511
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] JULY, AUGUST, SEPTEMBER 1955. SECTION II. Jan. 1956. Decl. Jan. 17, 1957. 186p. Contract W-31-109-eng-38. $1.10(OTS).

Experimental Boiling Water Reactor. The boiling water corrosion of Zircaloy-2 clad stainless steel thimbles is shown. Designs of the EBWR control rod and drive mechanism are given. The deceleration curve for the drive shaft at 600-psig operation is given. The prototype test facility for the control mechanisms is also shown. Calculated heat generation and thermal stresses in the pressure vessel wall for each type of thermal shielding are tabulated. The mechanical arrangement of the thermal shielding is given. The pressure vessel containment structure is described in detail, and calculated impact loading stresses are summarized. A redwood, steel, and celotex blast shield for the pressure vessel is also shown. Fast and thermal neutron flux calculations for the heavy water-reflected EBWR in the adjusted shield are summarized. The physical arrangement of this shielding is also described. An emergency boric acid injection system to reduce the reactor core activity below criticality was designed. The reactor water purification system is discussed. A schematic flow diagram of the air drying and fluid recovery system is given. Estimated flows in this system are also given, and the operation of the system is briefly discussed. A schematic plan view is shown of the EBWR power plant and control areas. A simplified line diagram is also given of the EBWR electrical system. The schematic for the emergency air-lock doors for the EBWR is shown. Supporting and Alternate Design Research and Development. The design of a boiling D_2O reactor is briefly discussed. Data given on boiling heat transfer studies include the ratio of steam-to-liquid velocity as a function of the variable inlet temperature, power, flow, and pressure. The analytical and experimental studies made of the 600 psi, unrestricted, single-channel circulation loop (Armadilla) are described, and preliminary data are plotted. These plots show modified Martinelli-Nelson factors as a function of slip ratio and exit and local void fraction. Data from single channel forced and natural circulation boiling density tests at 150 psig are summarized. Slip ratios are also plotted as a function of inlet water velocity at 150 and 250 psig, indicating that there is no significant pressure effect on the slip ratio pattern in going from 150 to 250 psig. The corrosion of Zircaloy 2-clad, U–Zr fuel plates in 550°F degassed water. The corrosion behavior of Al-clad, natural U plates is compared to that of EBWR reference alloy plates. The corrosion of Al–Ni alloy in 500°F H_2O is also shown. Corrosion data are also reported for austenitic stainless steel, and its weight changes are plotted. Corrosion and irradiation tests were continued on urania–thoria bodies. Several methods were tested for the detection and location of ruptured fuel elements in the core of a reactor. These included comparisons of fission product activities in various sampling devices and of γ activities in H_2O samples. It was, however, concluded that neither of these methods would be useful. A proposed alternate control rod for the EBWR is described and discussed. Progress in the development of boron–stainless steel alloy is discussed. Boron–stainless steel corrosion tests are summarized. Results of experiments on the removal of boric acid using Amberlite IRA-400 resin are summarized. Experimental Breeder Reactor I. The conversion ratio was measured in the second U fuel loading of the EBR-I. U^{238} capture distributions and U^{235} fission distributions in the reactor are shown. The conversion ratio with the cup about $\frac{1}{4}$ in. below its uppermost position was found to be 1.03 ± 0.05. Activation cross section ratios

at various locations in the outer blanket of the EBR were surveyed. Experimental and theoretical values are compared. Experimental Breeder Reactor II. The dissipation of simulated decay heat generated in the EBR-II core subassembly is shown. Water pressure drop tests were conducted on the complete EBR-II core subassembly, and the results of the tests are summarized. Calculations of hold-down forces in the core are also given. Irradiation tests were made on EBR-II fuel elements. EBR-II Working Model. A general description is given of the control rod and the cover raising and rotating mechanisms. A diagram of the fuel unloading system is given. Performance characteristics are given for a model tank pump in Na at 370°C. Supporting and Alternate Design Research and Development. The composition, grain size, and appearance of Mo–U and Zr–U alloys, which were thermal cycled in NaK are shown. Equipment is shown for the induction heating of the Na test loop, and the induction heating time for this loop is shown. Reactor Safety and Kinetics. Machine solutions of the kinetics equation are described in some detail. Analytical solutions of kinetics equations are also discussed. These methods were used to investigate the oscillatory behavior of the kinetic system for a reactor with a lifetime of 4×10^{-8} sec, and resultant calculations are plotted. The solution to the general problem of obtaining temperature distributions in two-region cylindrical geometry for an exponentially rising heat source in the central region is also given. The time variation of stored heat in a Na-cooled U rod during transient is also calculated, and the fraction of heat remaining in the fuel rod is shown as a function of the reciprocal period of the heat source. Fuel and Structural Materials. The failure in the tube assembly for the MTR loop water-to-water heat exchanger is discussed. Instrumentation. The pulse height distribution of fission signals in B^{10} and Al–U–Al neutron-sensitized photomultiplier tubes is shown. (C.H.)

978 ANL-5568(Del.)

Argonne National Lab., Lemont, Ill. and Nuclear Metals, Inc., Cambridge, Mass.

THE FABRICATION OF PROTOTYPE FUEL ELEMENTS FOR THE EXPERIMENTAL BOILING WATER REACTOR AND THE EXPERIMENTAL BREEDER REACTOR. Yearly Report: July 1, 1954 to June 30, 1955. H. F. Sawyer, P. Loewenstein, W. C. Paynton, and P. Corzine. May 1956. Decl. Mar. 14, 1957. 41p. Contracts W-31-109-eng-38 and AT(30-1)-1565. $0.40(OTS).

The methods used for producing prototype fuel elements for the Experimental Boiling Water and Experimental Breeder Reactors are described. Foundry and fabrication techniques and problems are discussed; the results of product evaluation are given. Extrusion constants for several U alloys and for Zircaloy-2 are included. Thermal expansion and transformation characteristics for the U-5 w/o Zr-1.5 w/o Nb alloy are given. (auth)

979 APAE-Memo-5

Alco Products, Inc., Schenectady, N. Y.

CALCULATION OF CONTROL ROD WORTH IN APPR-1. H. W. Giesler and J. G. Gallagher. Dec. 23, 1955. Decl. Apr. 1, 1957. 17p. Contract AT-(11-1)-318. $3.30(ph OTS); $2.40(mf OTS).

Seven control rods will be used in the APPR-1 instead of five as originally planned. Critical experiments evaluating the rod worth for the APPR-1 have been performed at ORNL, and calculations have been carried out by ALCO interpreting these experimental results. The calculations indicate that the capability of any four of the original five rods to shut down the cold reactor at mid-life is marginal. The calculations are summarized in this memorandum. (auth)

980 APAE-Memo-13

Alco Products, Inc., Schenectady, N. Y.

APPR-1 AXIAL NON-UNIFORM BURN UP—INITIAL STUDIES. F. B. Fairbanks. Mar. 28, 1956. Decl. Mar. 9, 1957. 14p. Contract AT-(11-1)-318. $4.80(ph OTS); $2.70(mf OTS).

The control rod position in relation to reactor lifetime and the relative concentrations of U^{235} and B^{10} as a function of U burned were found by a modified 2-group theory using the adjusted age and a straight 2-group theory without fast multiplication or resonance capture. Results indicate that at 15% average burnup the flux shape and reactivity do not vary appreciably from that given by the uniform analysis. (F.S.)

981 APAE-Memo-15

Alco Products, Inc., Schenectady, N. Y.

COMPARISON OF BORON AND HAFNIUM IN APPR CONTROL RODS. W. R. Johnson. Apr. 26, 1956. Decl. Feb. 22, 1957. 9p. Contract AT(11-1)-318. $1.80(ph OTS); $1.80(mf OTS).

A comparison of B and Hf for use in APPR control rods shows that it is possible to substitute Hf plate. Increased wall thickness of Hf will result in a smaller water volume, but reduction of the moderating center will not decrease the effectiveness of Hf rods. Burnup of Hf results in only small change of the initial blackness. Since neutron absorption in most of the Hf isotopes are (n,γ) reactions there is no heat generation in Hf rod. (W.L.H.)

982 APAE-Memo-43

Advanced Scientific Techniques Research Associates, Milford, Conn.

TEMPERATURE DISTRIBUTION AND THERMAL STRESS IN REACTOR CORE APPR-1. W. P. Berggren and H. R. Kroeger. Feb. 15, 1956. Decl. Mar. 12, 1957. 16p. For Alco Products, Inc., Contract AT(11-1)-318. (ASTRA-415E3.1). $3.30(ph OTS); $2.40(mf OTS).

983 BAW-2(Del.)

Babcock and Wilcox Co. [Atomic Energy Div., Akron, Ohio.]

LIQUID METAL FUEL REACTOR; TECHNICAL FEASIBILITY REPORT. June 30, 1955. Decl. with deletions Feb. 21, 1957. 465p. [For Brookhaven National Lab.] Contract AT-30-2-GEN-16, Subcontract No. S-262. $2.30(OTS).

The feasibility of the Liquid Metal Fuel Reactor (LMFR) proposed by Brookhaven National Laboratory as a source of electric power was thoroughly investigated. The reference design is based on an LMFR using U dissolved in Bi as the fuel and a thorium bismuthide in Bi slurry as the blanket. Detailed estimates of power costs are presented for electrical power stations from 226 to 904 Mw net electrical capability. (C.W.H.)

984 BNL-142(Del.)

Brookhaven National Lab., Upton, N. Y.

REACTOR SCIENCE AND ENGINEERING DEPARTMENT PROGRESS REPORT FOR JULY 1–SEPTEMBER 30, 1951. Decl. with deletions Feb. 14, 1957. 30p. Contract AT-30-2-gen-16. $4.80(ph OTS); $2.70(mf OTS).

Tests are reported on a liquid U–Bi continuously processed reactor and a normal water moderated and cooled reactor with 1% removal of U^{235} Zr-clad fuel rods for a Pu power producer. Removal of U tracer from ThF_4 with F is tabulated. Oxidation rates for irradiated graphite are studied. A general review is given the work of testing reactor components. Studies are reported on various liquid fuel systems for the liquid fuel reactors. (J.E.D.)

985 BNL-164(Del.)

Brookhaven National Lab., Upton, N. Y.

PROGRESS REPORT OF THE REACTOR SCIENCE AND

ENGINEERING DEPARTMENT [FOR] OCTOBER 1–DECEMBER 31, 1951. Clarke Williams. Decl. with deletions Feb. 14, 1957. 28p. $4.80(ph OTS); $2.70(mf OTS).

Liquid Metal Fuel Reactor component development continued with design of circulating loops, both in and out of pile, for testing fuel corrosion, fuel processing, and blanket behavior. The effect of decreased moderation on the outer part of the core and on the heat exchanger was studied. Processing of blanket and fuel was investigated, and data on the kinetics of the blanket regeneration cycle are given. Tube and shell-type heat exchangers are compared. Some calculations on a normal water reactor are given. Other studies reported on include: effect of spherical voids on gamma attenuation in water, two-column system for light end decontamination of UF_6, and creep corrosion and thermal cycling of BNL reactor fuel elements. (T.R.H.)

986 BNL-176(Del.)

Brookhaven National Lab., Upton, N. Y.

REACTOR SCIENCE AND ENGINEERING DEPARTMENT PROGRESS REPORT FOR JANUARY 1–MARCH 31, 1952. Clarke Williams. Decl. with deletions Feb. 13, 1957. 34p. Contract [AT-30-2-gen-16]. $6.30(ph OTS); $3.00 (mf OTS).

This report includes all the work on the liquid metal fuel reactor and normal water enriched uranium reactor component test project. In addition some studies on the present BNL research reactor, on shielding, BrF_3 extractive distillation, metallurgy, and a portion of the ultimate disposal treatment of Purex wastes are included. (auth)

987 BNL-309

Brookhaven National Lab., Upton, N. Y.

QUARTERLY PROGRESS REPORT [FOR] MAY 16–AUGUST 15, 1954. Decl. Apr. 2, 1957. 47p. $0.45(OTS).

Experimental measurements were continued for H_2O-slightly enriched U lattices. Thermal utilization factors and resonance escape probabilities were determined for 1.15, 1.027, and 1.30% enrichment and several H_2O-to-U volume ratios. The value of B^2 and k_∞ previously reported for H_2O-moderated natural-U lattices has been redetermined as a result of a more accurate measurement of the relaxation length. The new values are $B^2 = -3.48 \times 10^{-4}$ cm^2, and $k_\infty = 0.989$. Numerical solutions from UNIVAC calculations on the kinetics of a circulating fuel reactor have given information on the temperature limits as a function of reactor length. A proposed flowsheet for LMFR fuel processing is presented. The design provides for continuous removal of rare earth fission products. A program has been initiated to investigate methods for eliminating undesirable heating in underground liquid storage waste tanks. Data on the mass transfer of Fe by liquid Bi were obtained as a function of U, Zr, and Mg additions. Further investigations on the effect of dissolved Zr in liquid Bi indicate that the presence of this element has a marked effect on the precipitation and rate of solution of Fe in the Bi. Experiments on the oxidation of graphite show that the rate, when exposed to a γ field, is about the same as that with no radiation. The oxidation rate with pile irradiation is about 7 times as great as either of the other cases. The transmission of neutrons and γ rays through cylindrical air ducts in reactor shielding has been measured for a variety of geometric configurations and reactor operating conditions. (For preceding period see BNL-297.) (C.H.)

988 BNL-316(Del.)

Brookhaven National Lab., Upton, N. Y.

NUCLEAR ENGINEERING DEPARTMENT QUARTERLY PROGRESS REPORT [FOR] AUGUST 16–NOVEMBER 15, 1954. Decl. with deletions Feb. 12, 1957. 48p. $7.80(ph OTS); $3.30(mf OTS).

Buckling measurements reported in the last quarter have been extended to 1.027% enriched rods of 0.387-in. diameter. Preliminary results were obtained in an experiment to observe the variation in the fast fission factor with buckling. A number of LMFR kinetic equations were reduced to an infinite set of integro-differential equations involving the quantity $\phi_j(t)$. Corrosion studies were completed on a series of waste mixtures in contact with 1020 steel under boiling conditions. The feasibility of adsorbing volatilized RuO_4 on stainless steel was investigated. A preliminary phase diagram of the ternary Th–Pb–Bi system is presented. Settling experiments on Th–Bi dispersions in Bi and Pb–Bi were performed. Settling is normal in Bi at 400 and 500°C, and in Pb–Bi at 200 and 400°C. At 550°C in Pb–Bi, the particles would not settle out. Thermal pulsing of a Th–Bi dispersion to 700°C (21 cycles) produced a grain growth from 80 to 120μ without the formation of platelets. Diffusion of Pa (from irradiation of Th_3Bi_5) from solid Th_3Bi_5 to the liquid is very slow at 400, 500, and 600°C. The performance of a number of liquid-metal natural circulation loops is summarized. These loops are of Croloy, containing U–Bi, and U–Bi–Mg alloy. Preliminary data on the solubility of Cr in Bi are presented. The thermal annealing of reactor graphite at 350°C showed a 40 to 50% recovery of incremental growth produced by irradiation of samples after a first annealing. Graphite simultaneously oxidized and irradiated continues to lose weight at approximately 8 to 10 times the rate of unirradiated graphite. Work on the transmission of neutrons and γ rays through holes and ducts in water shields was continued. (C.H.)

989 BNL-380

Brookhaven National Lab., Upton, N. Y.

PROGRESS REPORT [FOR] MAY 16–NOVEMBER 15, 1955. Decl. Apr. 8, 1957. 74p. $0.55(OTS).

Development studies for the Liquid Metal Fuel Reactor are reported in reactor physics, chemistry, and chemical engineering, and metallurgy. (T.R.H.)

990 BNL-3009

Brookhaven National Lab., Upton, N. Y.

LMFR PROGRESS LETTER FOR SEPTEMBER 1956. F. T. Miles. Oct. 24, 1956. Decl. Mar. 25, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

The distribution of U between the carefully purified ternary eutectic and Bi was determined at 500°C. Data concerning the status of the loops are tabulated. (B.J.H.)

991 CF-3582

[Chicago. Univ.] Metallurgical Lab.

DESIGN OF THE FAST NEUTRON BREEDER TEST PILE. PART II. Report for January–June 1946. Aug 7, 1946. Decl. Feb. 15, 1957. 122p. Contract W-7401-eng-37. $0.60(OTS).

Part I was issued as CF-3414 (deleted version: TID-10085).

992 CF-50-9-37

[Oak Ridge National Lab., Tenn.]

REVIEW OF ORNL HOMOGENEOUS REACTOR EXPERIMENT BY THE REACTOR SAFEGUARD COMMITTEE. C. E. Larson. Sept. 11, 1950. Decl. Feb. 15, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Pre-operation safety studies for the Homogeneous Reactor Experiment are discussed. (C.H.)

993 CF-51-10-110

[Oak Ridge National Lab., Tenn.]

NUCLEAR CALCULATIONS FOR HOMOGENEOUS REACTORS PRODUCING U-233. S. Visner. Oct. 22, 1951. Decl. Feb. 14, 1957. 54p. $0.50(OTS).

As an aid in evaluating the economics and engineering feasibility of large-scale homogeneous reactors for produc-

ing power and U^{233}, nuclear calculations are made for highly enriched spherical reactors of both the one-region and two-region types. The two-region reactors investigated have a central core surrounded by ThO_2-D_2O slurry in a blanket. Separating the core and blanket is a Zr shell whose absorption is taken into account in the calculations. In the core is a solution of UO_2SO_4 in D_2O, the fuel isotope being U^{235} for the converters and U^{233} for the breeders. Calculations are made with various relative power levels in the blanket due to the fission of the U^{233} produced. Among the other parameters studied are thickness of the core shell, temperatures of the core and blanket, concentration of Th in the blanket, and blanket thickness. A study is made of the behavior of the reactors as a function of operation time in order to take into account the build-up of higher isotopes of U. Single-region reactors with no blanket or reflector are also investigated, but not as extensively. The fuel solution, UO_2SO_4 in D_2O, is considered homogeneously mixed with ThO_2 in various concentrations. (C.H.)

994 CF-52-2-76
Oak Ridge National Lab., Tenn.
ESTIMATED COPPER CONCENTRATIONS REQUIRED FOR OPERATION OF HOMOGENEOUS REACTORS. H. F. McDuffie. Feb. 11, 1952. Decl. Apr. 2, 1957. 4p. Contract [W-7405-eng-26]. $0.25(OTS).

995 CF-52-8-220(Del.)
Oak Ridge School of Reactor Technology, Tenn.
MTR TYPE POWER PRODUCER FOR A REMOTE LOCATION. Reactor Design and Feasibility Problem. T. M. Hallman, N. Hilvety, R. W. Klecker, C. D. McKereghan, W. R. Pearce, T. Stern, R. M. Summerhays, and G. M. White. Aug. 20, 1952. Decl. with deletions May 7, 1957. 164p. $24.30(ph OTS); $7.50(mf OTS).

A power system is described whereby 7500 kw of electrical power is produced by two identical nuclear reactors operating in parallel to supply a common boiler and, in turn, to drive a low-pressure steam turbine–generator set. This system is especially designed for power generation in an isolated location where fuel transportation difficulties make other power sources impractical. In this report, a new method of shim control is suggested for the reactors which reduces the otherwise high initial excess reactivity required for long periods of operation. This method involves the addition of a predetermined amount of nuclear poison to the fuel which is burned out at a rate comparable to that of the fissionable material. This tends to maintain the multiplication factor of the reactor at a nearly constant value. (auth)

996 CF-52-8-230
Oak Ridge School of Reactor Technology, Tenn.
PRELIMINARY DESIGN CALCULATIONS FOR A FAST U^{238} CONVERTER. Reactor Design and Feasibility Problem. W. E. Davis, C. H. Bergsland, Harold Brammer, J. H. Kinginger, J. C. Nance, W. E. Schortmann, and P. H. Wackman. Aug. 20, 1952. Decl. Mar. 14, 1957. 142p. $22.80(ph OTS); $7.20(mf OTS).

A study was made on the nuclear behavior of a fast reactor based on a liquid U – Fe fuel system, with particular emphasis on the region of low U^{235} enrichment (10–20%). An enriched core–U^{238} blanket system was briefly investigated also. The U–Fe system was found to be operable from a nuclear standpoint but would present extremely difficult engineering problems because of the high temperatures involved. (M.H.R.)

997 CF-53-8-146
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE FAST-NEUTRON SPECTRUM OF THE BULK SHIELDING REACTOR USING NUCLEAR PLATES. M. P. Haydon, E. B. Johnson, and J. L. Meem. Aug. 31, 1953. Decl. Feb. 13, 1957. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A nuclear plate camera was developed at the Bulk Shielding Facility to measure the fast-neutron spectrum of the reactor. A proton-recoil type fast-neutron spectrometer was used to measure the reactor spectra as attenuated by the BeO reflector and various amounts of H_2O. Since data can be obtained much more quickly with the spectrometer, no extensive use of nuclear plates is contemplated. However, the nuclear plate method is of established reliability, and it was considered desirable to check the spectrometer at at lease one point. It was found that the total integrated flux as measured by the two method differs by approximately 11.3%, which is within the statistical accuracy of the two instruments. (auth)

998 CF-53-10-39
Oak Ridge National Lab., Tenn.
THE OBSERVATION OF URANIUM PEROXIDE PRECIPITATION IN THE HOMOGENEOUS REACTOR. E. L. Compere. Oct. 6, 1953. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $0.25(OTS).

Analyses of the UO_4 precipitated in the Homogeneous Reactor fuel solution have resulted in the following conclusions: (1) UO_4 is formed if the reactor power level exceeds values characterized by temperature and the catalytic characteristics of the soup; (2) Solid UO_4 disappears slowly at temperatures below 75°C; (3) Solid UO_4 may be removed from the soup by thermal decomposition by heating without power operation to temperatures of 100 to 120°C for 2 hrs or less. (M.P.G.)

999 CF-54-1-45
Oak Ridge National Lab., Tenn.
CRITICAL MASS CALCULATIONS FOR A PROPOSED REBUILT HRE. M. Tobias. Jan. 4, 1954. Decl. Feb. 7, 1957. 11p. Contract [W-7405-eng-26]. $0.25(OTS).

Two-group critical mass calculations were made for several reactor sizes based on the general configuration of a proposed rebuilt Homogeneous Reactor Experiment. Design criteria are summarized. Results of the calculations are presented in tabular form. (C.H.)

1000 CF-54-4-81
Oak Ridge National Lab., Tenn.
POSSIBLE HAZARD IN HRT START-UP. Sidney Visner. Apr. 8, 1954. Decl. Apr. 2, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The possibility of an accident to the reactor, due to the direct pumping of the concentrated fuel into the empty core tank, is examined. (B.J.H.)

1001 CF-54-5-229
Oak Ridge National Lab., Tenn.
CRITERIA FOR USE IN THE ESTIMATION OF THE COST OF SHIPPING CONTAMINATED REACTOR FUELS. F. T. Binford. May 20, 1954. Decl. Apr. 2, 1957. 25p. [Contract W-7405-eng-26]. $0.35(OTS).

A procedure for estimating the quantities of fission products present in discharged fuel from a reactor which has been operated at steady state for an extended period and has decayed for a specified time is given. The procedure is extended to include the fission products present in the blanket of a breeder reactor. The shield weight required to permit shipment of daily batches of this material can be computed by conventional methods, using the values obtained for the quantities of fission products present. Curves showing approximate shipping costs via rail express and via motor freight for various typical distances are shown. It is recommended that containers for unprocessed fuel and blanket material be identical with those used for the shipment of processed fuel. (auth)

1002 CF-55-1-78(Del.)
Oak Ridge National Lab., Tenn.
SUMMARY OF BOILING SYSTEMS SESSION AT THE REACTOR HEAT TRANSFER INFORMATION MEETING AT BROOKHAVEN NATIONAL LABORATORY, OCTOBER 18–19, 1954. P. C. Zmola. Jan. 13, 1955. Decl. with deletions Mar. 11, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Both favorable and unfavorable features of boiling in reactors are briefly discussed. (L.M.T.)

1003 CF-55-2-101
Oak Ridge National Lab., Tenn.
ALTERNATE CORE AND REFLECTOR DESIGNS FOR THE APPR. J. G. Gallagher. Feb. 15, 1955. Decl. Feb. 23, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

1004 CF-55-3-183
Oak Ridge National Lab., Tenn.
EFFECT OF CORE DESIGN ON THE REACTIVITY CHANGE DURING THE LIFETIME OF THE APPR. J. G. Gallagher. Mar. 29, 1955. Decl. Feb. 23, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

1005 CF-55-6-31
Oak Ridge National Lab., Tenn.
SPECIFICATIONS FOR STAINLESS STEEL MTR IRRADIATION TEST ELEMENT. IRRADIATION REQUEST ORNL-17. R. J. Beaver. June 2, 1955. Decl. Mar. 29, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A testing program to evaluate performance of an experimental prototype stainless steel APPR fuel element under reactor conditions is outlined. The fabrication and inspection procedure for the test component are described. (T.R.H.)

1006 CF-55-6-165
Oak Ridge National Lab., Tenn.
CONTROL ROD ABSORBER SECTION SPECIFICATIONS FOR APPR CRITICAL EXPERIMENT. H. G. Blosser. June 28, 1955. Decl. Feb. 23, 1957. 4p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Design data, specifications, and drawings are presented for control rod absorber plates for the Army Package Power Reactor critical experiment. (L.T.W.)

1007 CF-55-9-120
Oak Ridge National Lab., Tenn.
POWER AND TEMPERATURE OF TBR FOLLOWING INTERRUPTION OF NORMAL OPERATION. M. W. Rosenthal. Sept. 27, 1955. Decl. Feb. 19, 1957. 39p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Calculations were performed to determine the behavior of the TBR under certain conditions involving interruption of normal operation. The situations considered are sudden stoppage of circulation in the core circuit, sudden cessation of heat removal in the heat exchanger, and circulating pump failure. The first two situations result in a rapid, hazardous temperature rise; no danger appears in the third condition. (D.E.B.)

1008 CF-56-2-115
Oak Ridge National Lab., Tenn.
PRELIMINARY RESULTS OF APPR CRITICAL EXPERIMENTS. PART III. D. V. P. Williams. Feb. 8, 1956. Decl. Feb. 22, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Measurements were made of the value of the Army Package Power Reactor control rods in various locations in terms of the mass of U^{235} required to maintain criticality, and the results are tabulated. The critical mass was also obtained as a function of the boron loading. (B.J.H.)

1009 CF-56-3-178(Del.)
Oak Ridge National Lab., Tenn.
TWO-GROUP PERTURBATION CALCULATION OF CHANGE IN MULTIPLICATION CONSTANT WITH MASS FOR THE ARE. Joel Bengston. Mar. 12, 1956. Decl. with deletions Feb. 20, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Within the framework of two-group theory, first order perturbation theory has been used to obtain expressions for the change in multiplication constant due to small changes in uranium concentration in a reactor in which the percentage of non-thermal fissions is appreciable. The method has been applied to a two-region model of the ARE, for which it appears that the simple bare reactor formula gives results in good agreement with the more general expression for a reflected reactor. Since the weakest point in a two-group critical mass calculation is the evaluation of the fast constants, the critical mass was varied by changing the epithermal uranium cross sections, and the sensitivity to increased mass, measured by $R = \Delta k_{eff}/k_{eff}/\Delta m/m$, for $k_{eff} \simeq 1$, was found to decrease strongly, but approximately linearly, with increase in critical mass. (auth)

1010 CF-56-4-51
Oak Ridge National Lab., Tenn.
DECAY AND STORAGE OF IRRADIATED FUEL. J. W. Ullmann and E. D. Arnold. April 9, 1956. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Reasons for storage to allow decay of irradiated fuel elements and possible methods of storage are given. The effects on storage and inventory costs of fuel element composition, plant size, element geometry, reactor type, methods of irradiation and recycle, and type of metallurgical handling are discussed. Estimates and comparisons are included for the decay for several typical fuels. The special problems associated with thorium fuel elements are considered. (auth)

1011 CF-56-4-75
Oak Ridge National Lab., Tenn.
RESULTS OF PRELIMINARY NUCLEAR CALCULATIONS FOR A HOMOGENEOUS RESEARCH REACTOR. C. W. Nestor, Jr. Apr. 12, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Critical mass, fuel concentration, power, maximum power density, and wall power density were obtained for bare spherical reactors having diameters of 8, 10, 11, and 12 ft, operating at 20, 100, 200, and 300°C. (L.T.W.)

1012 CF-56-4-99
Oak Ridge National Lab., Tenn.
PROCEDURE FOR TESTING REFRIGERATION SYSTEM AND COLD TRAPS—HRT TEST VII A. N. W. Curtis. Apr. 14, 1956. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Procedures are outlined for cleaning, filling, testing, and operating the Homogeneous Reactor Test refrigeration system and cold traps. (W.L.H.)

1013 CF-56-4-105
Oak Ridge National Lab., Tenn.
HRT LEAK DETECTION SYSTEM. J. E. Kuster. Apr. 20, 1956. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

All HRT process piping and equipment is contained in a large tank and flanged connections with stainless steel ring gaskets are used where needed to permit the removal of values and items of equipment. Underwater remote maintenance is to be employed and special provisions are required for indicating and locating leaks at all mechanical joints in the process system. Each joint is monitored and a signal is given when a leak occurs. The valve operator

stems are sealed with stainless steel bellows and a means of detecting a leak in the bellows has been included. (auth)

1014 CF-56-4-134
Oak Ridge National Lab., Tenn.
HRT COOLING-WATER SYSTEM. HRT Engineering Test Procedures VI A-2, 3, 4, 5; B-1, a, b; B-2 and B-3. R. E. Brooksbank. Apr. 18, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Experimental procedures for testing the HRT cooling-water system and its components are presented. The cooling-water system is a closed system consisting of three parallel circuits fed by a single circulating pump fed from a forced draft cooling tower and supply reservoir. In the event of radioactive contamination of the cooling-water system the tower supply lines are equipped with block valves that confine the solutions to the reactor cell, and after shutdown heat may be removed by discharging contaminated cooling water to the cell. Failure of the circulating pump will actuate a valve that will permit convection cooling to take place within a closed circuit. (auth)

1015 CF-56-8-117
Oak Ridge National Lab., Tenn.
HRT-CP INSTRUMENTATION DESIGN MANUAL. R. K. Adams. Aug. 17, 1956. 25p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The installation and design of the HRT-Core Processing Plant are described in detail so that operation and maintenance of the plant may be made more effective. Process instruments will be described fully, but auxiliary instrumentation such as pressure gauges, rotameters, etc. will not be mentioned. Where detailed questions arise as to electrical circuitry or instrument connections, the drawings and specification sheets should be consulted. (auth)

1016 CF-56-8-166
Oak Ridge National Lab., Tenn.
EXAMINATION OF TITANIUM PULSE FEEDER DIAPHRAGM FAILURE. T. M. Kegley, Jr. Aug. 24, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A Ti pulse feeder diaphragm failed by cracking after 450 hr service in the Homogeneous Reactor Test mock-up. The diaphragm, which was made of MST Grade III Ti, was vibrated at 78 cpm at 40 to 50°C. One side of the diaphragm was exposed to demineralized H_2O and the other side was exposed to 10g U/l UO_2SO_4 solution. The main corrosion product was TiO_2. (W.L.H.)

1017 CL-FD-15
Clinton Labs., Oak Ridge, Tenn.
A HOMOGENEOUS HIGH TEMPERATURE POWER PILE UTILIZING GRAPHITE WHICH HAS BEEN IMPREGNATED WITH URANIUM. Farrington Daniels. Aug. 23, 1946. Decl. Feb. 14, 1957. 2p. $1.80(ph OTS); $1.80(mf OTS).

1018 DC-51-10-24
[General Electric Co. Aircraft Nuclear Propulsion Project, Cincinnati.]
THERMAL NEUTRON DIFFUSION AND KINETIC EQUATIONS CORRESPONDING TO THERMAL FERMI AGE EQUATION CORRECTED FOR FAST FISSION. Fritz W. Mezger. Oct. 25, 1951. Decl. Mar. 15, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

Thermal reactors that have a significant fraction of fissioning occurring just above thermal energies are studied. First, the thermal neutron diffusion equation is corrected for fast fission effects. The resulting diffusion equation is shown to be equivalent to the Fermi Age equation corresponding to the same physical assumptions. Next, the one delayed neutron group kinetic equations for the corrected thermal flux are given, and the life time of the thermal neutron is corrected by the average slowing down time. (auth)

1019 HW-24327
Hanford Works, Richland, Wash.
A STUDY OF THE RADIATION BURST IN THE HANFORD HOMOGENEOUS REACTOR. B. R. Leonard, Jr. May 2, 1952. Decl. Feb. 28, 1957. 30p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

A partially full (93%) spherical 20-in. diam. reactor using plutonium nitrate fuel was accidentally brought to a prompt critical condition. The energy released in the ensuing reaction was measured to be about 3 megawatt seconds from the temperature rise and the increased beta activity of the fuel. The behavior of the system during the incident has been analyzed. It is shown that the incident resulted from the rapid withdrawal of the safety rod from the system. The dominant feature of the incident was the rapid expansion of the fuel to attain a more favorable geometry until the sphere was full. The net effect of the expansion, however, was to kill the reaction. The time behavior of the incident has been analyzed in two ways. In the first place, an energy release of 3 megawatt seconds was assumed and from the estimated rates of withdrawal of the safety rod the minimum times for the incident to take place and the shortest periods reached are calculated. The times are found to be greater than about 200 milliseconds and the shortest periods possible are about 10 milliseconds. In the second place, the possible mechanisms of stopping the reaction are evaluated. The incident has then been treated on the basis of a calculation by Fuchs (LA-596). This analysis shows that the incident occurred in less than 0.5 seconds and that power levels the order of 5×10^7 watts were encountered. The energy release predicted here is in agreement with that measured. (auth)

1020 HW-32791
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
SUMMARY REPORT OF REACTOR HAZARDS FOR THE PROTOTYPE PHYSICAL CONSTANTS TESTING REACTOR. D. J. Donahue, D. D. Lanning, W. A. Horning, and R. L. Dickeman. Sept. 16, 1954. Decl. Apr. 4, 1957. 116p. Contract W-31-109-eng-52. $18.30(ph OTS); $6.00(mf OTS).

The safety aspects of the design and operation of the Prototype Physical Constants Testing Reactor (PCTR) are discussed in some detail. This reactor is essentially a seven-foot cube of graphite with a large hole at its center, about $2' \times 2' \times 3'$. It is made critical with uranium enriched to at least 92 per cent in U^{235} and runs normally at a power of about one watt. Sections of various graphite–U lattices are inserted into the large hole at the center of the reactor, and the effects of these test lattices on the reactivity of the PCTR are noted. There are two mechanical safety systems on the reactor; eight shutter-type control rods, which are driven to the closed position by springs in about 0.25 seconds, and four cadmium disks which fall freely into the reactor under the force of gravity in about 0.5 seconds. These 12 mechanical safeties have individual driving and electromagnetic release mechanisms. The electromagnets are controlled by flux monitoring through five electronic channels, two of which are proportional counter channels for subcritical and post-critical monitoring, while the remaining three are ion chamber channels for post-critical and level monitoring. If all of these mechanical devices fail, the hollow fuel elements in the driver rods will melt and collapse causing an increase in self-shielding of the U^{235}. When this occurs, the reactor will lose about six per cent in $\Delta k/k$. The col-

lapsing of fuel elements occurs quickly enough to shut down the reactor before the fuel vaporizes, after the instantaneous introduction of anything less than three per cent Δk/k. Finally, if an excursion continues until the fuel elements in the driver rods vaporize, short-lived fission products from about 3×10^{17} fissions will be released to the atmosphere. An analysis of the distribution of these fission products indicates that little damage is to be expected outside the area in which the reactor is located. (auth)

1021 HW-33655

[Hanford Atomic Products Operation, Richland, Wash.]
WATER TREATMENT SYSTEM FOR INTERMEDIATE CAPACITY DUAL PURPOSE REACTOR PLANT—RDS-10 AND RDS-11. W. L. Pearl. Nov. 26, 1954. Decl. May 8, 1957. 13p. Contract [W-31-109-eng-52]. $0.30(OTS).

Water quality requirements, water treatment, cost aspects, and site plan are outlined for a water treatment facility in a dual-purpose reactor plant. (C.W.H.)

1022 HW-34388(Del.)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
A PRELIMINARY STUDY OF THE ECONOMICAL INCENTIVES OF IN-PILE BOILING AS APPLIED TO DUAL MARKET REACTOR PLANTS. W. C. A. Woods. Jan. 10, 1955. Decl. with deletions May 8, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

A study was performed to determine the economic incentives associated with in-pile boiling, with the explicit assumption that this method of operation was technically feasible. A non-boiling reactor plant was used as a comparison, and the net electric generation was held constant at 400 Mw. The economic effect of variations in specific power from 47 to 32 kw/ft., and steam qualities of 10 to 60% were studied. The results showed that, under the conditions used, the unit power cost during single purpose operation would be in the range 5.9 to 6.1 mils/kw-hr at the generator bus-bar, compared to an estimated 6.6 mils/kw-hr. for the non-boiling plant. These gains are realized under conditions of relatively high specific powers, greater than 32 kw/ft., relatively modest bulk outlet steam quality (greater than 20%), and high-pressure steam (1000 psia or higher) used directly in the turbines. During the dual-market period, the power rate is about the same. An increase of 0.2 mils/kw-hr. is indicated when the specific power is lowered. Capital investment savings of perhaps four to five million dollars may be possible, but savings of this magnitude have little effect on the over-all power cost. The power cost savings stated above are almost entirely fuel savings, a direct result of the higher thermal efficiency inherent in the in-pile boiling cases. (auth)

1023 HW-37641(Del.)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
A GRAPHITE SUPERHEATING REACTOR. R. M. Fryar. July 1, 1955. Decl. with deletions Feb. 22, 1957. 41p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

A design study has been made of a graphite superheating reactor, a graphite moderated reactor with a heat rating of 305 Mw which produces superheated steam. It is designed for the production of plutonium and electrical power or for the production of electrical power only. The fuel elements for dual-purpose operation are U metal, while for single-purpose action, they are uranium oxide. The fuel elements are cooled by a 7-pass system in which preheated water enters the bottom of the first pass and is discharged as superheated steam at the top of the seventh pass. The reactor control consists of shim control by the gas system, vertical gas-cooled control rods, vertical safety rods, and a ball safety system. Rough cost estimates indicate that the capital cost of the reactor should be comparable to other reactors of similar heat output. (D.E.B.)

1024 IDO-16001

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
REPORT ON MTR FUEL ASSEMBLY DATA CORRELATION. F. P. Vance and M. N. Hudson. Nov. 8, 1951. Decl. Mar. 30, 1957. 19p. $3.30(ph OTS); $2.40(mf OTS).

The limits of uncertainty inherent in computations of SF material content of MTR fuel assemblies were calculated. The major source of uncertainty is systematic error in the chemical analysis for U. (T.R.H.)

1025 IDO-16009

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
REVIEW OF SHIELDING CALCULATIONS FOR IRRADIATED FUEL ASSEMBLIES IN THE MTR CANAL. W. L. Morris and D. T. Bray. Apr. 1, 1952. Decl. Feb. 26, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

The canal shielding calculations for the MTR are reviewed. It is concluded that the present amount of water in the canal should provide adequate shielding. (B.J.H.)

1026 IDO-16082

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
NUCLEAR CONSTANTS FOR THE MTR WITH 168-GRAM FUEL ELEMENTS AND 131-GRAM FUEL-BEARING SHIM RODS. M. W. Holm, J. W. Webster, and H. L. McMurry. May 8, 1953. Decl. Mar. 30, 1957. 32p. Contract AT(10-1)-205. $0.30(OTS).

1027 IDO-16153(Del.)

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
TECHNICAL QUARTERLY REPORT FOR FOURTH QUARTER, 1953. J. R. Huffman. Feb. 1, 1954. Decl. with deletions Feb. 26, 1957. 37p. Contract AT(10-1)-205. $6.30(ph OTS); $3.00(mf OTS).

The flux surveys of the MTR were essentially completed, thereby making available to the experimenters fairly complete knowledge of thermal fluxes and, to a lesser extent, fast flux values throughout the reactor. Verification of the validity of the use of Co^{60} as a flux monitor was an important part of this program. The cross sections program was considerably advanced, yielding publication of data on the cross section of Zr and Cl in the range from 0.02 to 50 ev. These data were obtained on the crystal spectrometer. The cross section value of about 6.2 b at 10 ev is in disagreement with previously published values of 9 b. An extension of the application of scintillation techniques to the solution of problems at the MTR was made. The design of a new facility at the MTR for the study of organic fluids in high-intensity γ-ray fields was completed. (auth)

1028 IDO-16327

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
LOCAL THERMAL FLUX DISTRIBUTION IN SPERT III FUEL ELEMENTS BY THE SPHERICAL HARMONICS METHOD. C. W. Reich. Aug. 20, 1956. 28p. Contract AT(10-1)-205. (PTR-103). $0.25(OTS).

The local thermal flux distribution in water moderated SPERT III stainless steel fuel elements was calculated by the spherical harmonics method, using P_1, P_3, and P_5 approximations. From the flux thus determined, spatial averages of the macroscopic fission cross section, the macroscopic absorption cross section, and the thermal

utilization were calculated and compared with the values obtained by treating the fuel-moderator system as homogeneous. (auth)

1029 IDO-16346
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
CAPACITY OF THE MTR COOLING SYSTEM. J. H. Rainwater and V. A. Walker. Feb. 13, 1956. 17p. Contract AT(10-1)-205. (PTR-42). $3.30(ph OTS); $2.40(mf OTS).

A study of the Materials Testing Reactor cooling system was made to estimate the capacity of the system as a function of reactor inlet water temperature. Calculations and tests are summarized which indicate it is not feasible to increase reactor power to 60 Mw until cooling system modifications are made. (D.E.B.)

1030 KAPL-48
Knolls Atomic Power Lab., Schenectady, N. Y.
A REVIEW OF THE BREEDING PROBLEM IN NUCLEAR REACTORS. H. Brooks. Mar. 29, 1948. Decl. Mar. 18, 1957. 34p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf OTS).

The characteristics of power breeder reactors are discussed in extremely general terms. (B.J.H.)

1031 KAPL-131
Knolls Atomic Power Lab., Schenectady, N. Y.
PRELIMINARY INVESTIGATION OF A FAST REACTOR. R. Ehrlich. Mar. 15, 1949. Decl. Mar. 5, 1957. 20p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Brief multi-group analyses were made of the size and power distribution of reactors having no slowing-down material other than Na and structural material in the core. Investigations were made of assemblies with a 6-cm Be reflector and an infinite U blanket, a 12-cm MgO reflector and an infinite U blanket, or an infinite U reflector. Results of the analyses are tabulated. (B.J.H.)

1032 KAPL-238(Del. 2)
Knolls Atomic Power Lab., Schenectady, N. Y.
TECHNICAL FEASIBILITY REPORT FOR THE KAPL WEST MILTON AREA REACTOR. Feb. 14, 1950. Decl. with deletions Mar. 8, 1957. 280p. Contract W-31-109-Eng-52. $43.20(ph OTS); $11.10(mf OTS).

This feasibility report is presented under the following topics: objectives of the program; description of reactor system; criticality considerations; nuclear components—design, materials, fabrication, assembly, and testing; reactor structure; reactor controls; liquid metal heat transfer systems; power conversion; loading, transfer and storage facilities; shielding; processing fuel and breeder rods; experimental program and facilities; servicing reactor components; special waste handling facilities; and plan for reactor operation. (L.M.T.)

1033 KAPL-277
Knolls Atomic Power Lab., Schenectady, N. Y.
REACTOR ENGINEERING, SECTION IV [OF] PROGRESS REPORT NO. 40 [FOR] NOVEMBER 1–30, 1949. Dec. 23, 1949. Decl. May 13, 1957. 35p. Contract W-31-109-Eng-52. $9.30(ph OTS); $3.60(mf OTS).

1034 KAPL-334
Knolls Atomic Power Lab., Schenectady, N. Y.
SECULAR EQUATIONS FOR BREEDING. Harvey Brooks. Apr. 28, 1950. Decl. Mar. 27, 1957. 20p. Contract W-31-109-Eng-52. $0.25(OTS).

Equations are given for an elementary analysis of breeding, simultaneous breeding and conversion, and the influence of higher isotopes on breeding. A numerical example is included. (M.H.R.)

1035 KAPL-1136
Knolls Atomic Power Lab., Schenectady, N. Y.
STUDY OF A BOILING REACTOR STEAM PLANT FOR CENTRAL STATION POWER. J. W. Weil, E. L. Zebroski, and T. M. Snyder. May 17, 1954. Decl. Feb. 27, 1957. 36p. Contract W-31-109-Eng-52. $0.35(OTS).

Fundamental considerations in the design of boiling reactors are discussed. It is shown that two types of stability are available to steam-generating reactors, the low-steam-content type of stability corresponding to reactors of the Borax design. The other type of stability occurs in the high-steam-content region and is designated as "empty tube" stability. A reactor based on the "empty tube" principle is described and various aspects of its design are considered. The lattice consists of a rod-type fuel element surrounded by a low-density mixture of saturated steam and water. This mixture is confined in a process tube. The space between adjoining process tubes is filled with low-temperature water. The advantages of such a reactor design are discussed with respect to application to central station generation of electricity. (auth)

1036 KAPL-M-CAB-1(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
FUEL ELEMENTS FOR HETEROGENEOUS REACTORS. C. A. Bruch, W. M. Cashin, and H. H. Hirsch. May 17, 1955. Decl. Mar. 11, 1957. 33p. Contract W-31-109-eng-52. $6.30(ph OTS); $3.00(mf OTS).

Problems inherent in fuel element design are reviewed, and a method of attack on the problem of producing better fuel elements is included. (J.H.M.)

1037 KAPL-M-CLM-1
Knolls Atomic Power Lab., Schenectady, N. Y.
THERMAL TEST REACTOR CRITICALITY CONSIDERATIONS. C. L. McClelland. Mar. 26, 1951. Decl. Mar. 12, 1957. 14p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Results of experiments to test the contribution to pile reactivity of the various components of the design of the Thermal Test Reactor (TTR) are reported. Minor modifications of the TTR are discussed, with an estimate of the potential reduction in critical mass for the revised TTR to be assembled at the KAPL site. (L.T.W.)

1038 KAPL-M-CM-2
[Knolls Atomic Power Lab., Schenectady, N. Y.]
SUMMARY OF EXPERIMENTAL WORK RELATING TO SCHENECTADY POWER PILE. C. Mannal. Dec. 15, 1947. Decl. Mar. 13, 1957. 32p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf OTS).

A brief summary of experimental work done on the KAPL Intermediate Power Breeder is given. The material is divided into sections on applied physics, coolants, control, mechanisms, instruments, physics, chemistry and metallurgy, each page summarizing one experiment. (T.R.H.)

1039 KAPL-M-CVM-3
Knolls Atomic Power Lab., Schenectady, N. Y.
THE GAS PRESSURE-ACTUATED DELIVERY OF A LIQUID POISON CONTAINED IN A POISON FUSE INTO THE REACTOR CORE FOLLOWING FUSE TRIGGER OPERATION. C. V. Moore. June 18, 1951. Decl. Mar. 29, 1957. 16p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Dimensionless design curves are derived and presented which relate the major discharge characteristics of a gas pressure-actuated liquid poison safety fuse—such as the time required for discharge, the volume of poison delivered and the initial pressure required. These curves, being di-

mensionless, may readily be applied to any specific case. For the case of a mercury poison a discharge time of 10 milliseconds is obtained when the fuse is loaded to 45 percent of the theoretical maximum volume of poison with an initial gas pressure of 1920 psi. The analysis used here neglects fluid resistance effects. As a result, the stroke times given are probably low by a factor of 2 or 3. If more accurate results are required, these may be obtained by more rigorous analysis and by model tests. It appears from these calculations that almost any discharge times may be obtained, provided the appropriate compromises are made in poison volume and initial pressure. (auth)

1040 KAPL-M-DRM-2

[Knolls Atomic Power Lab., Schenectady, N. Y.]
CRITICAL MASSES OF BARE SPHERICAL THERMAL PILES WITH BERYLLIUM MODERATOR AND U^{235} FUEL. D. R. Miller. July 24, 1947. Decl. Feb. 11, 1957. 6p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Calculated data are presented on critical masses of bare spherical piles with pure U^{235} fuel and Be moderator as a function of pile radius for various volume fractions of various coolants. The calculations are based on one-velocity theory with a correction made for loss of neutrons by leakage during slowing down. Absorption of neutrons of energy above thermal was neglected. (auth)

1041 KAPL-M-EDB-1

Knolls Atomic Power Lab., Schenectady, N. Y.
HYDROGEN EVOLUTION RATES IN SIR MARK A PRIMARY SHIELD TANK. Edward L. Brady. Aug. 22, 1955. Decl. Mar. 14, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Hydrogen evolution rates in the Mark A Submarine Intermediate Reactor primary shield tank are reviewed and interpreted. It is concluded that part, but not all, of the wide fluctuations in the measured rate is due to the behavior of the hydrogen measuring instrument; possible explanations for the rest of the fluctuations are offered. The apparent general downward trend in the observed rates is probably real and is attributed to the inhibition of radiation-induced decomposition by hydrogen dissolved in the shield tank, produced by corrosion reactions. (auth)

1042 KAPL-M-EEB-2

Knolls Atomic Power Lab., Schenectady, N. Y.
PROBLEM NUMBER 2, DEVELOPMENT AND EVALUATION OF BORON CARBIDE SHIELDING MATERIALS. REPORT NUMBER 1, THERMAL CYCLING OF SIR PROTOTYPE CONTROL RODS. E. E. Baldwin. June 17, 1952. Decl. Mar. 9, 1957. 12p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Three Submarine Intermediate Reactor (SIR) prototype control rods filled with vibrating mix boron carbide were thermally cycled 500 times in sodium between 200 and 540°C to determine the effect of cyclic temperature changes on the structural stability of these members. The only dimensional change which occurred was an increase in the bow of the rods, due most probably to relief of residual stresses in the stainless steel tubing. Gas evolved from the boron carbide during the test. (auth)

1043 KAPL-M-FHC-3

Knolls Atomic Power Lab., Schenectady, N. Y.
FAST EFFECT IN A HOMOGENEOUS H_2O–U-238 REACTOR. F. H. Clark, Jr. Apr. 25, 1952. Decl. Mar. 29, 1957. 8p. Contract W-30-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A computation is made of the number of fast U^{238} fissions which occur for each fission of any kind in a reactor composed of a homogeneous mixture of H_2O, U^{238}, and a small amount of U^{235}. Results of the calculations are tabulated. (C.H.)

1044 KAPL-M-GD-4

Knolls Atomic Power Lab., Schenectady, N. Y.
BLANKET FISSIONS IN SAPL V. Gerhard Dessauer. Aug. 26, 1949. Decl. Mar. 9, 1957. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The number of blanket fissions, both U^{235} and U^{238}, were calculated as a percent of core fissions. Blanket fission distributions were plotted. (D.E.B.)

1045 KAPL-M-HAB-2

[Knolls Atomic Power Lab., Schenectady, N. Y.]
SUMMARY OF BLANKET CALCULATIONS. H. A. Bethe. July 9, 1947. Decl. Mar. 29, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

1046 KAPL-M-HB-26

Knolls Atomic Power Lab., Schenectady, N. Y.
NOTES ON THE FEASIBILITY OF A DUAL-PURPOSE REACTOR. H. Brooks. Aug. 26, 1949. Decl. Mar. 8, 1957. 25p. Contract [W-31-109-eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Uncertainty regarding the value of intermediate reactors for breeding throws increased emphasis on the need for maximum flexibility of neutron spectrum in the proposed KAPL reactor. The natural alternative of a fast or "semi-fast" reactor appears to require an excessive amount of fuel, perhaps 130 kg, and owing to limitations on total power by the Safeguards Committee, leads to a specific power but little in excess of CP-4. On the other hand, studies of a fast reactor have indicated that in many respects the fast reactor fuel element is superior to that developed specifically for the intermediate reactor. Since it appears probable that future power breeders may be fast, one task of the KAPL reactor should be to test fast fuel elements at higher specific power than is possible in a true fast pile at present. The dual-purpose reactor employs the same basic tube matrix and the same reflector and blanket structure as the present intermediate reactor. Most sections of the matrix will be filled partly with fast pile fuel elements and partly with solid Be rods which fill the complete matrix tube without any annulus for coolant. The reflector and blanket sections of the fuel elements will be standard as designed for the fast pile. The reflector sections of the Be rods will also be solid Be, while the blanket sections may be solid Fe. It is impossible to have blanket sections above and below the Be rods, because there is no passage for coolant. The relative number of Be and fuel rods in the core will be adjusted to give an over-all Be–U^{235} ratio of about 40 to 1. However, since the fuel rods contain Na and the Be tubes do not, the fuel rods will have about the same reactivity value in the intermediate spectrum as the Be rods, and the two are thus interchangeable within limits. This interchangeability will permit a substantial variation of spectrum. (auth)

1047 KAPL-M-HB-31

Knolls Atomic Power Lab., Schenectady, N. Y.
EFFECTIVENESS OF BLANKET FISSIONS. H. Brooks and H. A. Bethe. Apr. 28, 1950. Decl. Mar. 12, 1957. 8p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Calculations are made to show the influence of a primary U^{235} or U^{239} fission in the blanket on the conversion or breeding ratio. (W.L.H.)

1048 KAPL-M-HES-1

Knolls Atomic Power Lab., Schenectady, N. Y.
THE CAPACITY OF THE WMA STANDBY COOLING SYSTEM SHOULD BE REDUCED FROM 2½% TO 1% OF OVERLOAD REACTOR CAPACITY. H. E. Stone. Apr. 4,

1950. Decl. Mar. 9, 1957. 16p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

1049 KAPL-M-HRP-1
Knolls Atomic Power Lab., Schenectady, N. Y.
S2G PLUG DRIVE EQUIPMENT. Herbert R. Pagano. May 11, 1956. 10p. Contract [W-31-109-Eng-32]. $1.80 (ph OTS); $1.80(mf OTS).

The Submarine Intermediate Reactor (S2G) plug drive equipment consists of the concentric plug drive mechanism, the eccentric plug drive mechanism, the emergency plug drive, the concentric position transmitter, the eccentric position transmitter, and the plug drive console. The service line handling facility is also included. (L.T.W.)

1050 KAPL-M-IB-8
Knolls Atomic Power Lab., Schenectady, N. Y.
A MULTIGROUP PROCEDURE FOR CALCULATING BARE REACTORS. Ira Bornstein. Dec. 23, 1952. Decl. Feb. 26, 1957. 11p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

A summary is given for a multigroup procedured used for calculating non-hydrogeneous intermediate bare reactors. (F.S.)

1051 KAPL-M-JDS-1
Knolls Atomic Power Lab., Schenectady, N. Y.
A COMPARATIVE ANALYSIS OF THE LIQUID METAL HEAT TRANSFER SYSTEMS FOR WMA. J. D. Selby. Apr. 20, 1949. Decl. Feb. 26, 1957. 84p. Contract W-31-109-Eng-52. $13.80(ph OTS); $4.80(mf OTS).

Design criteria are presented for heat transfer units for WMA (Intermediate Power Breeder). A comparative analysis was made of system size for use in setting up the final specifications for the heat transfer units. The reactor was designed to generate steam at a temperature of 550°F and 465 psia, including 90° superheat potentially capable of an efficiency of 28% in electrical output, and to use molten Na as the primary coolant. Factors investigated include the effects of heat exchanger size, steam pressure, and liquid metal flow rate on load rating and top reactor temperature. Specifications for the heat transfer system are outlined. No system recommendations are presented. (C.H.)

1052 KAPL-M-JEZ-1
[Knolls Atomic Power Lab., Schenectady, N. Y.]
RECENT DEVELOPMENTS ON THE STANDBY COOLANT SYSTEM FOR THE WEST MILTON REACTOR. John E. Zerbe. June 20, 1949. Decl. Mar. 27, 1957. 12p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The work done previously on the Standby Coolant System was reviewed in the light of more recent reactor designs and heat transfer data. It has been found that if a NaK–Na heat exchanger consisting of $16-1\frac{1}{4}$" OD 60 mil wall stainless steel tubes is used, the thermal stress is reduced to approximately 22,000 psi when $2\frac{1}{2}\%$ of 30 Mw is transferred under steady state conditions. Further calculations based on dissipating $2\frac{1}{2}\%$ of 30 Mw have shown that the thermal stress is not influenced noticeably by the temperature rise in the NaK system. This is for both 80-20 (eutectic) NaK and 50-50 NaK. However, assuming the NaK-air exchanger is placed at a height of 55 feet the factor of safety for the various temperature rises in the NaK loops is different. Thus, for 80-20 NaK, if ΔT_{NaK} equals 200°F, calculations show that the exchanger must be at least 40 feet in the stack; for ΔT_{NaK} 150° the exchanger must be located at about 86 feet. If 50-50 NaK were used, the factor of safety would be greater than with 80-20 NaK. (auth)

1053 KAPL-M-JKD-7
Knolls Atomic Power Lab., Schenectady, N. Y.
ACTIVATION OF HOMOGENEOUS REACTOR COOLANTS. J. K. Davidson. Apr. 10, 1953. Decl. Mar. 8, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A mathematical analysis is presented of the activation of homogeneous reactor coolants. (W.L.H.)

1054 KAPL-M-KAK-3
Knolls Atomic Power Lab., Schenectady, N. Y.
GENERAL PIN EQUATION. C. R. Stahl and K. A. Kesselring. [1948?]. Decl. Mar. 7, 1957. 23p. $6.30(ph OTS); $3.00(mf OTS).

An equation is derived for the relationship existent among all constants and variables experienced in a mechanical analysis of a pin-type, Ni or stainless steel clad reactor fuel element. Curves are included for sample selections of fuel pin parameter variations for probable design, materials, and environments. (T.W.S.)

1055 KAPL-M-MSN-6
Knolls Atomic Power Lab., Schenectady, N. Y.
TEMPERATURE COEFFICIENT OF PPA. M. S. Nelkin. Aug. 3, 1956. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The expression for the thermal utilization derived in MSN-4 is differentiated with respect to temperature to obtain a positive temperature coefficient of $(2.0 \pm 0.2\ \phi/°F$ for the ANARPA loading of the PPA. (auth)

1056 KAPL-M-MSP-1
Knolls Atomic Power Lab., Schenectady, N. Y.
REACTOR SHUTDOWN BY NUCLEATE BOILING. PART I. SURFACE AREA IN NUCLEATE BOILING. M. S. Pendergast, Jr. June 6, 1956. 58p. Contract W-31-109-eng-52. $9.30(ph OTS); $3.60(mf OTS).

The problem to be solved is that of determining the conditions under which a reactor such as S3G/S4G, which accidentally has reached a fast period, may be controlled by the use of nucleate boiling, without either rod burnout, bulk boiling, or metallurgic phase change of the fuel. This report deals primarily with determining the surface area of a rod which is in nucleate boiling, and is therefore only a first step in the over-all problem. (auth)

1057 KAPL-M-PGL-1
Knolls Atomic Power Lab., Schenectady, N. Y.
PROGRAM FOR ALTERNATE COEXTRUSION PROCESS FOR REFERENCE SAR FUEL ELEMENTS. P. G. Lozier. Sept. 24, 1956. 6p. Contract W-31-109-Eng-52. $1.80 (ph OTS); $1.80(mf OTS).

A preliminary study was made of procedures for coextruding Submarine Advanced Reactor fuel elements with integral end closures. It was found that 5 element assemblies can be coextruded in tandem. A proposed program is outlined for the study of extrusion variables. (C.H.)

1058 KAPL-M-PLH-2
Knolls Atomic Power Lab., Schenectady, N. Y.
CALCULATION OF LOW-ENERGY SELF-SHIELDING FACTORS FOR A SPECIFIC REACTOR GEOMETRY. P. L. Hofmann. Jan. 16, 1952. Decl. Mar. 9, 1957. 12p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Procedures are outlined for calculating various low-energy self-shielding factors applicable to an "SIR-type" geometry. (L.T.W.)

1059 KAPL-M-PLH-9
Knolls Atomic Power Lab., Schenectady, N. Y.
CALCULATION OF SOME DANGER COEFFICIENTS BY MEANS OF PERTURBATION THEORY. P. L. Hofmann. Dec. 30, 1952. Decl. Mar. 9, 1957. 24p. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Danger coefficients for various Submarine Intermediate Reactor constituents have been calculated by means of

perturbation theory. The results are compared to those obtained experimentally from critical configurations. Formulas and certain numerical integrals used in the calculations are tabulated. (auth)

1060 KAPL-M-RE-1
Knolls Atomic Power Lab., Schenectady, N. Y.
MULTI-GROUP RESULTS ON BLANKETED ASSEMBLIES. R. Ehrlich. Sept. 19, 1947. Decl. Feb. 7, 1957. 16p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The multigroup method for determining the critical size of a reactor was applied to assemblies consisting of a core, Be reflector, and U^{238} blanket. Six energy groups were used in the calculations, and complete results are given in both graphical and tabular form. (B.J.H.)

1061 KAPL-M-RE-3
[Knolls Atomic Power Lab., Schenectady, N. Y.]
FURTHER MULTI-GROUP RESULTS WITH MODERATED BLANKETS. R. Ehrlich. Mar. 22, 1948. Decl. Mar. 13, 1957. 38p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf OTS).

The results of four multi-group calculations are reported. The first is an extension of calculations contained in Memo-RE-1, being for the same core and reflector, but with a finite 15 cm Be–U blanket (pile 1). It is thus the calculation best simulating the 50-1 Be–U pile with 25% Na by volume. The three pile 2 calculations estimate the neutron properties of a 40-1 Be–U assembly with 40% Na by volume. Several schemes are considered. (D.E.B.)

1062 KAPL-M-RGK-4
Knolls Atomic Power Lab., Schenectady, N. Y.
TEMPERATURE ANALYSIS OF THE WMA REACTOR SLOW BREEDER REGION. R. G. Kennison. May 26, 1950. Decl. Mar. 9, 1957. 12p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Temperature distribution calculations for the WMA reactor slow breeder region are presented for both a Be matrix and a tube matrix arrangement. It was found that the maximum rod temperature in the innermost row for the Be matrix arrangement reaches 1160°F. Forced cooling of this row allows the maximum temperature to be reduced to 1020°F. With the tube matrix arrangement, the maximum rod temperature in the innermost row was found to be 1040°F. Both arrangements are acceptable from a heat transfer standpoint. (auth)

1063 KAPL-M-RGK-8
Knolls Atomic Power Lab., Schenectady, N. Y.
TEMPERATURE DISTRIBUTIONS IN THE SIR ROTATING RELOADING PLUGS — MARK III DESIGN. R. G. Kennison. Jan. 10, 1953. Decl. Mar. 11, 1957. 84p. Contract W-31-109-Eng-52. $13.80(ph OTS); $4.80(mf OTS).

The steady state and transient solutions made to determine the temperature distributions in the SIR rotating reloading plugs are presented. These are almost entirely analytical solutions as machine facilities to solve the problems were not available. They are applicable in most cases not only to the present plug design but also to other similar designs since quite general conditions were assumed for the analytical solutions. (auth)

1064 KAPL-M-RJF-1
Knolls Atomic Power Lab., Schenectady, N. Y.
HOLD-UP TIMES FOR SAFE FUEL ROD REMOVAL FROM THE SCHENECTADY REACTOR. R. J. Fritz. Aug. 19, 1948. Decl. Mar. 12, 1957. 13p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Upon removal from the Schenectady reactor, a fuel rod is lifted from the Na coolant into a He filled cavity above the reactor. Due to the difference in heat transfer coefficients for the two mediums, and due to continued heat generation in the rod, a problem of maintaining safe temperatures exists. The hold-up times, time between shutdown and rod removal, required for the maintenance of safe temperatures were calculated for reactors operating at 10 and 30 Mw. (D.E.B.)

1065 KAPL-M-RSM-3
Knolls Atomic Power Lab., Schenectady, N. Y.
ESTIMATED INTERNAL PRESSURES IN FAST OXIDE BREEDER FUEL ELEMENTS DUE TO FISSION GAS RELEASE. R. S. Miller. Dec. 5, 1956. Decl. Mar. 19, 1957. 19p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

An estimate of the increase in the internal pressure of the KAPL Plutonium Power Breeder Reactor fuel pins due to the accumulation of fission gases was made for ranges of specific power, fuel burnup, and fuel density. Families of curves are presented which show the variation of internal pressure with fuel diameter for various values of the parameters listed above. Allowable upper limits on fuel diameter have been determined for the various cases considered as a function of the volume fraction of cladding iron present in the core region. (auth)

1066 KAPL-M-RSS-3
Knolls Atomic Power Lab., Schenectady, N. Y.
EXPERIMENTAL DETERMINATION OF THE SPECTRA IN SEVERAL COLD WATER MODERATED ASSEMBLIES. R. S. Stone and R. E. Slovacek. Aug. 3, 1956. 26p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Neutron spectra were obtained for a series of subcritical fuel assemblies with light water as moderator. It is concluded from the experimental results that the observed spectra are in reasonable agreement with the Wigner-Wilkins thermal neutron spectra calculations. (L.T.W.)

1067 KAPL-M-RWL-3
Knolls Atomic Power Lab., Schenectady, N. Y.
RESULTS OF THE 360°, FULL SCALE, WATER FLOW TEST OF THE TOP OF THE KAPL REACTOR WITH HOLD-DOWN PLATE NUMBER 1. R. W. Lockhart and R. G. Kennison. May 16, 1949. Decl. Mar. 11, 1957. 19p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40 (mf OTS).

1068 KAPL-M-SMS-10
Knolls Atomic Power Lab., Schenectady, N. Y.
SOLUBILITY AND FLOW CHARACTERISTICS OF AMMONIUM PENTABORATE COLUMNS. E. L. Shirley and A. P. Walsh. Nov. 22, 1955. Decl. Mar. 14, 1957. 22p. S3G/S4G Project. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The proposal to use a column of ammonium pentaborate as the means for introducing a neutron absorbing nuclide into the Submarine Advanced Reactor (S3G/S4G) systems for chemical shutdown has required data on the solubility and flow characteristics of this salt when a stream of water is passed through the column. The test equipment is described, and the results of the solubility and pressure measurements are presented. (auth)

1069 KAPL-M-SWK-6
Knolls Atomic Power Lab., Schenectady, N. Y.
MEASUREMENT OF MIGRATION AREA AND INFINITE MULTIPLICATION IN HIGHLY ENRICHED, CRITICAL REACTORS. S. W. Kitchen and D. R. Bach. June 14, 1956. 13p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

In principle it is possible to measure the migration area by critical measurements near zero buckling. If the product of the migration area and the buckling is less than 0.1, the true migration area can be obtained directly from the

slope of the infinite multiplication factor plotted against the buckling. If M^2B^2 is greater than 0.1, this slope may differ from the true migration area by more than ten percent. In practice it is not often practicable to make critical measurements with reactors using highly enriched fuel having M^2B^2 less than 0.1. The true migration area and the infinite multiplication factor can be determined, however, by fitting a polynomial in B^2 to critical measurements of the thermal utilization at large as well as at intermediate values of B^2. This procedure will also define the minimum number of groups in a multigroup expression that correctly describes the reactor compositions studied. In the absence of measurements at large B^2, it is possible to define a migration area by fitting a two-group expression to the logarithm of a calculated thermal utilization plotted against the measured buckling. This curve can be used to predict criticality in the range of the measured buckling to within plus or minus a dollar of reactivity. (auth)

1070 KAPL-M-WAN-6

Knolls Atomic Power Lab., Schenectady, N. Y.

A REACTOR PHYSICS PARAMETER SURVEY OF ZIRCONIUM–WATER, ROD-CONTROLLED, REACTORS. W. A. Northrop, S. Pearlstein, T. F. Ruane, and I. A. Rouillard. May 6, 1955. Decl. Mar. 4, 1957. 38p. Contract W-31-109-Eng-52. $6.30(ph OTS); $3.00(mf OTS).

The results of an extensive reactor physics parameter survey are presented. The purpose of the survey was to determine how several important quantities vary with reactor composition and dimensions. Fuel loading, temperature coefficient, axial power factor, radial power factor, and water gap peaking factor were among the quantities determined. (auth)

1071 KAPL-M-WJD-2

Knolls Atomic Power Lab., Schenectady, N. Y.

A SURVEY OF BURNABLE POISONS TO BE USED IN THE STRUCTURAL MATERIAL OF A CORE. W. J. Dodson. May 5, 1954. Decl. Feb. 7, 1957. 9p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Since it would be desirable to incorporate a burnable poison into the structural material of a reactor core, a survey was carried out to see if a suitable material could be found. In performing the calculations, it was first assumed that the reactivity at the end of 2000 hours is the same as at startup. Under this restriction, it was found that the burnable poison cross section must be greater than the fuel cross section. The only isotopes found that might be used and that give a ΔK excursion less than that of B^{10} are Ir^{191}, Hg^{199}, and Dy^{164}. If after 2000 hours operation $k_\infty(K_\infty(1))$ is allowed to drop below the value of K_∞ at startup ($K_\infty(O)$) then elements with a cross section less than that of the fuel may be used as a burnable poison. However, for reasonable values of $K_\infty(O)$, $K_\infty(1)$, and initial fuel loadings the cross section a burnable poison must have is still about one-half the fuel cross section. This is much higher than the cross section of elements usually used as alloying materials in steel and so it is felt that these materials cannot be used as burnable poisons. (auth)

1072 KAPL-M-WRK-3(Rev.)(Del.)

Knolls Atomic Power Lab., Schenectady, N. Y.

CONFERENCE ON THE EXPERIMENTAL BREEDER REACTOR AT THE ARGONNE NATIONAL LABORATORY ON APRIL 25, 1952. M. Davis, H. Hurwitz, Jr., W. R. Kanne, T. M. Synder, H. E. Stevens, Jr., A. D. Tevebaugh, and C. E. Weber. [nd]. Decl. with deletions Mar. 12, 1957. 17p. Contract W-31-109-ENG-52. $3.30(ph OTS); $2.40 (mf OTS).

Information obtained from and experiments planned for the EBR are discussed by ANL and KAPL personnel. (L.M.T.)

1073 LA-1525

Los Alamos Scientific Lab., N. Mex.

MATERIAL REPLACEMENT EXPERIMENTS: THEORY AND MEASUREMENTS FOR THE LADY GODIVA ASSEMBLY. Gordon E. Hansen and Clifford Maier. Apr. 23, 1953. Decl. Apr. 1, 1957. 58p. Contract W-7405-Eng-36. $9.30(ph OTS); $3.60(mf OTS).

The perturbation theory for material replacement experiments in a bare assembly is given through second order, thus permitting corrections for sample size. Computed flux and adjoint distribution functions are tabulated for the Lady Godiva assembly, enabling the observed danger coefficients for U^{238} and U^{235} to be compared with corresponding predicted values. Consistency of this data is checked by its use in three independent combinations, each yielding the same value for the effective fraction of delayed neutrons from fast fission. Reactivity contributions associated with inelastic scattering were calculated and comparison made between central danger-coefficient ratios and ratios for the Topsy assembly. Evaluation of transport cross sections by means of replacement measurements in Godiva is illustrated for the several elements C, Cu, and Au. (auth)

1074 LA-1548

Los Alamos Scientific Lab., N. Mex.

A SIMPLE METHOD OF CALCULATING CRITICAL MASSES OF PROTON MODERATED ASSEMBLIES. George I. Bell. May 1953. Decl. Apr. 1, 1957. 31p. Contract W-7405-eng-36. $0.35(OTS).

Semi-empirical modifications of age theory have been made to interpret the critical masses of proton moderated assemblies which may or may not have reflectors of various substances, in particular, water or steel. Critical masses of untamped water-moderated systems can be calculated by use of an appropriate age and linear extrapolation length. Assuming that all captures and fissions take place at thermal, a familiar expression can be set up which contains the age and extrapolation length which are not well determined by theory. An age value suggested by theory is taken, and ORNL criticality data (K343) are used to find the best extrapolation length. With these values the K343 critical masses can be computed to about ±5%, which is their estimated experimental uncertainty; this for H/U^{235} atomic ratios between 43 and 755. A theoretical discussion of effects which the simple "theory" of neglects is presented. It is indicated that several of these effects compensate to make a simple theory more useful than one would believe at first sight. To take into account reflectors of water, water shielded by cadmium, or steel of various thicknesses, it proves sufficient to alter the extrapolation length appropriately. For moderators which contain appreciable HNO_3, H_3PO_4, or steel, parameters are given which enable one to compute the appropriate ages and extrapolation lengths. Critical masses are again calculated to within about ±5%. A simple mode generalization is suggested for large scale inhomogeneities in the fuel region which preserve a constant age throughout. The critical equations and numerical parameters are assembled in the appendix. (auth)

1075 LA-2029

Los Alamos Scientific Lab., N. Mex.

TIME BEHAVIOR OF GODIVA THROUGH PROMPT CRITICAL. T. F. Wimett, L. B. Engle, G. A. Graves, G. R. Keepin, Jr., and J. D. Orndoff. Apr. 1956. 36p. Decl. Apr. 2, 1957. 36p. Contract W-7405-eng-36. $0.40 (OTS).

Studies were made of the time behavior of a bare spherical U^{235} metal assembly (Godiva) following sudden increases in reactivity with essentially zero initial power level and no external neutron source. The highest final reactivity

attained under controlled conditions was about 10 cents above prompt critical. In the region above prompt critical the observed fission rate increased exponentially until the decreasing reactivity from thermal expansion caused it to pass through a bell-shaped maximum or "burst." Following this burst, the fission rate dropped to a low-amplitude "tail" sustained by delayed neutrons and finally terminated upon automatic scramming of the assembly. Burst amplitudes corresponding to instantaneous power levels of 10,000 Mw were observed with durations under a millisecond. Excellent agreement was found between measured reactor periods and calculations based on recent delayed neutron data. Over-all burst behavior also was found to agree with theory based on a space independent one-group model. (auth)

1076 LAMS-733(Del.)
[Los Alamos Scientific Lab., N. Mex.]
EFFECT OF TEMPERATURE AND REACTIVITY CHANGES IN OPERATION OF THE LOS ALAMOS PLUTONIUM REACTOR. David B. Hall and Jane H. Hall. May 28, 1948. Decl. with deletions Mar. 18, 1957. 40p. Contract [W-7405-eng-36]. $7.80(ph OTS); $3.30(mf OTS).

The operation of the Fast Reactor is considered in terms of normal equilibrium conditions and normal shut-down. The proposed loading, control rod adjustment and subsequent "floating" operation are discussed. Safety devices and interlocks are described. Temperature and reactivity changes are examined with respect to various system failures, phase changes, and "flashing" of the reactor. Slow changes due to faulty slug cooling are also considered. The calculations were initially based upon 10 kw operation. Performance tests of the mercury system now indicate that 20-kw operation may be feasible. (D.E.B.)

1077 LRL-56(Del.)
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
REACTOR DESIGN EVALUATION PROGRAM. Quarterly Progress Report No. 1 [for] January Through March 1953. Oct. 1953 Decl. with deletions Feb. 28, 1957. 12p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

This is the first quarterly progress report on the Reactor Design Evaluation Program. The requested objective is to make technical and economic evaluations of various reactor systems on a comparable basis, for production of power or for production of weapons-grade Pu, and for comparing the merits of reactor systems, MTA process, and incremental ore procurement for maintaining production of fissionable material during emergency periods. Work done to date has been largely collection of basic technical and economic data, including "ground rules" or uniform basis for comparison. This has included making assumptions as to the time and locations of possible nuclear power plants and collection of guidance data as to economics of steam pressure and temperature in conventional power plants. Evaluation studies were initiated for published power-only reactor proposals and for a new MTA electronuclear Pu production process design. (auth)

1078 LWS-24934 (Del.)
California Research and Development Co., Livermore, Calif.
CONCEPTUAL DESIGN OF THE ANL-FCR. S. E. Warner, W. B. Myers, J. C. Ekvall, and B. Lund. July 8, 1953. Decl. with deletions Feb. 26, 1957. 36p. $6.30(ph OTS); $3.00(mf OTS).

An initial design and economic analysis of the Fast Converter Reactor, a forerunner of the EBR, is presented. This design employs stainless steel-clad plate type fuel and breeding elements, liquid Na cooling, and has a power rating of 500 Mw. General reactor structure and operation are discussed, and fuel element assembly, installation, and removal details are given. (D.E.B.)

1079 MonN-389
Clinton Labs., Oak Ridge, Tenn.
CRITICAL MASS AND CRITICAL SIZE OF THERMAL GAS COOLED PILES. H. E. Stevens, Jr. Sept. 25, 1947. Decl. Feb. 26, 1957. 72p. $0.40(OTS).

A theoretical treatment of the critical mass and critical size of gas cooled thermal reactors surrounded by reflectors is presented. Many graphs based upon this approach are included for cylindrical reactors having Be, BeO, and C moderators, having various fuels and enrichment, and having many average temperatures. Curves of power output, pumping power, neutron economy, weight of materials, and specific power can be superposed upon the graphs, and an optimum design may be chosen according to the relative importance of the various factors. (L.T.W.)

1080 MonP-206(Del.)
[Clinton Labs., Oak Ridge, Tenn.]
PHYSICAL DATA ON NEW HIGH FLUX PILE. A. M. Weinberg. Nov. 21, 1946. Decl. with deletions Mar. 7, 1957. 33p. Contract W-35-058-eng-71. $6.30(ph OTS); $3.00(mf OTS).

Included are general discussions of engineering, physical, control, critical, and heat transfer data on a proposed H_2O-cooled and moderated reactor with a Be reflector. (B.J.H.)

1081 MTA-44
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
EFFECTIVE ONE ENERGY GROUP FISSION CROSS SECTION OF URANIUM-238 IN FAST REACTOR CORES. K. Bernstein. May 1, 1953. Decl. Apr. 3, 1957. 10p. Contract AT(11-1)-74. $0.25(OTS).

Approximate methods have been developed, using two energy group diffusion theory, to estimate the effective one energy group fission cross section of U^{238} for use in preliminary fast reactor core calculations. The method has been applied to plutonium–uranium and U^{235} enriched uranium fuel mixtures homogeneously mixed with structural material and coolant. Available experimental data from the EBR and critical assemblies at Los Alamos indicates a discrepancy of about ten per cent between the estimated and measured effective U^{238} fission cross sections. (auth)

1082 N-1145nd
Chicago. Univ. Metallurgical Lab.
FLEXIBILITY OF CARTRIDGES WITH JACKET SLIDING ON ONE SIDE ONLY. G. Young. Nov. 6, 1944. Decl. Mar. 22, 1957. 5p. (MUC-GY-22). $1.80(ph OTS); $1.80 (mf OTS).

Calculations were made to determine the flexibility of a tube when there is a restriction on the slippage between jacket and slug. (M.H.R.)

1083 N-2322
Argonne National Lab., Lemont, Ill.
"CANE" AND "BAMBOO" FUEL RODS. H. B. Fairchild. Aug. 7, 1946. Decl. Mar. 18, 1957. 4p. (ANL-OCS-31). $1.80(ph OTS); $1.80(mf OTS).

The advantages and disadvantages of tube-shaped fuel rods over previously considered designs are listed. Brief discussions are also given on the comparison of observed pressure drops of tube-rods with previous designs, prediction of pressure drops through tube-rod bundles, and dimensions of tube rods. (M.H.R.)

1084 NAA-SR-283
North American Aviation, Inc., Downey, Calif.
A CONCEPTUAL DESIGN OF A THORIUM–URANIUM

(233) POWER BREEDER REACTOR. J. O. Henrie and E. F. Weisner. Feb. 1, 1954. Decl. Mar. 1, 1957. 41p. Contract AT-11-1-GEN-8. $0.80(OTS).

A conceptual design study was performed for a Na cooled, graphite moderated, thermal power-breeder reactor utilizing the Th–U^{233} breeding cycle. Several aspects of the design of the system are considered but no attempt was made to supply all the details. It appears that the design presented is feasible and will allow the production of economic power as well as full utilization of Th resources. (auth)

1085 NAA-SR-1028

North American Aviation, Inc., Downey, Calif.

A SODIUM COOLED, GRAPHITE MODERATED, LOW ENRICHMENT URANIUM REACTOR FOR THE PRODUCTION OF USEFUL POWER. Edward F. Weisner, ed. Sept. 15, 1954. Decl. Mar. 4, 1957. 138p. Contract AT-11-1-GEN-8. $0.55(OTS).

A design study is presented for a sodium cooled, graphite moderated power reactor utilizing low enrichment uranium fuel. The design is characterized by dependence on existing technology and the use of standard, or nearly standard, components. The reactor has a nominal rating of 167 thermal megawatts, and a plant comprising three such reactors for a total output of 500 thermal megawatts is described. Sodium in a secondary, non-radioactive, circulation system carries the heat to a steam generator at 910°F and is returned at 420°F. Steam conditions at the turbine throttle are 600 psig and 825°F. Cost of the complete reactor power plant, consisting of the three reactors and one 150-megawatt turbogenerator, is estimated to be approximately $43,165,000. (auth)

1086 NAA-SR-1300

North American Aviation, Inc., Downey, Calif.

A SODIUM-GRAPHITE REACTOR STEAM-ELECTRIC STATION FOR 75 MEGAWATTS NET GENERATION. Edward F. Weisner and Warren M. Sybert. Mar. 22, 1955. Decl. Feb. 28, 1957. 65p. Contract AT(04-3)-49. $0.45(OTS).

The major design features, nuclear characteristics and performance data for a nuclear fueled central station power plant of 75,000 kw net capacity are presented. The heat source is a Na cooled graphite moderated reactor. The design of the reactor takes full advantage of the experience gained to date on the Sodium Reactor Experiment (SRE); the plant described here is a straightforward extension of the smaller experimental SRE, which is now under construction. The fuel elements are made up of rod clusters and the moderator is in the form of Zr canned graphite elements. The performance of the reactor has been based on conservative temperatures and coolant flow velocities which result in a plant with "built-in reserve." Thus, as experience is gained and anticipated improvements in reactor fuel elements and construction materials are proven, the performance of the plant can be increased accordingly. Two reactor designs are described, one for operation with slightly enriched U fuel elements and the other for operation with Th–U fuel elements. The associated heat exchangers, pumps, steam, and electrical generating equipment are identical for either reactor design. An analysis of turbine cycles describes the particular cycle chosen for initial operation and discusses a method by which modern central station performance can be initially obtained. The design and performance data which are required to enable reliable estimates of the plant construction and operating costs to be made are established. (auth)

1087 NAA-SR-1582

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1955. SECTION A. A. B. Martin, ed. SECTION B. J. C. Cochran, ed. Apr. 15, 1956. Decl. Apr. 8, 1957. 100p. Contract AT(04-3)-49. $0.60(OTS).

An analysis was made of the nuclear parameters for sodium graphite reactor lattices. These parameters include thermal utilization, macroscopic cross sections, thermal diffusion length, and neutron absorption. Results of all calculations are given in graphical form. Test fuel slugs for the SRE were cycled up to 500 times between 100 and 500°C at the rate of 2 cycles/hr. Results are tabulated. The centrifugal casting of U alloy fuel slugs is briefly evaluated. Results of the microscopic examination of the extruded Th-U breeder fuels are shown. The percent elongation of graphite due to the presence of Na is shown for various temperatures. Results of wear tests on graphite are also tabulated. The behavior of Zr in liquid Na was studied, and weight gains in Zr are summarized. Analog computer studies were continued, and data are included on the temperature effects of the response time of coolant channel Na outlet temperature thermocouples, the effects of continuous rod motion and pump speed changes on the outlet Na temperature and power, and the outlet temperature as a function of scram time. The critical evaluation of B–Ni rods is tabulated. The fuel rod assembly apparatus is described. Fuel rod development is discussed. Cyclograph traces of rods bonded with various Na–K alloys were recorded for rods at room temperature and heated to 450 and 600°F. The traces are indicative of uniform bonding. The moderator can fabrication and testing is also discussed. Tests were completed on Freeze Seal No. 2 for the 6-in. oval port Wedgeplug test valve at 450, 850, and 1250°F. The temperature gradients from the hot flange face to the end of the seal mechanism for various valve temperature conditions are shown. Sodium leak rates through the valve are tabulated. Progress in the development of a liquid Na level gage is briefly reported. The tubular heater experiment was completed, and the times to raise pipe temperatures from ambient to 350°F are tabulated. Designs for a 6-in. Na pump loop are described briefly. A one to 3.5 scale model of a SRE fuel element was constructed to study the effect of side drag on the element during insertion operations at those fuel channels located near the outlet of the upper plenum chamber. The calibration of the SRE fuel element orifices was studied. Control rod lead screw development is discussed. Development of the safety rod system is described. Core tank galling tests are summarized. Experiments to determine the effects of radiation on MoS_2 are described. Dose buildup factors for the concretes to be used in the reactor top shield are tabulated. Constants for the quadratic representation of the dose buildup factor and the capture γ rays from heavy concrete are also tabulated. The sodium pumps and service system are described. The rated cooling capacity of the SRE tetralin evaporative cooler was checked. Results of a study on the effects of NaK temperature on the H content of He are tabulated. (D.E.B,)

1088 NAA-SR-Memo-37

North American Aviation, Inc., Downey, Calif.

PROPOSED WATER BOILER NEUTRON SOURCE. E. R. Cohen, T. Fahrner, A. Biehl, and E. M. Kelly. May 3, 1951. Decl. Feb. 27, 1957. 11p. Contract AT-11-1-gen-8. $9.30(ph OTS); $3.60(mf OTS).

A preliminary design study of a water boiler reactor to be used as a neutron source for the NAA exponential program shows that the small size of the water boiler makes it an inexpensive and ideal producer of thermal neutrons. Because of the low power operation (30 to 100 milliwatts) there are no difficulties associated with the disposal of radioactive gases. (F.S.)

1089 NAA-SR-Memo-313
North American Aviation, Inc., [Downey, Calif.].
PROPOSED EXPERIMENTAL PHYSICS PROGRAM FOR THE CONVERTER REACTOR PROJECT. R. A. Laubenstein. May 5, 1952. Decl. Apr. 8, 1957. 10p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

A proposed mockup designed to supply data for a converter reactor to produce U^{233} from Th with the use of highly enriched U^{235} as fuel is discussed. The Th and U^{235} requirements and the cost estimation for this mockup are included in an appendix. (M.H.R.)

1090 NAA-SR-Memo-673
North American Aviation, Inc., [Downey, Calif.]
EFFECT OF REFLECTOR THICKNESS AND VARYING AMOUNTS OF ALUMINUM AND WATER ON THE REFLECTOR SAVINGS FOR THE PLUTONIUM PRODUCTION CORE OF THE UPR. Ralph Balent. Apr. 27, 1953. Decl. Feb. 28, 1957. 12p. Contract [AT-11-1-Gen-8]. $3.30 (ph OTS); $2.40(mf OTS).

In order to help determine the radial thickness and permissible amounts of poisons for the graphite reflector of the plutonium production core of the UPR, a few two-group two-region criticality calculations were made. A curve of reflector savings, in per cent of an infinite reflector, as a function of thickness for pure graphite reflectors is given. A curve of savings, in per cent of an infinite pure graphite reflector, as a function of amount of aluminum and water content is given. The results show that a two foot graphite reflector will yield a reflector savings of 85% of that obtainable with an infinite reflector. Five percent aluminum reduces the reflector savings of a pure graphite reflector by approximately 10% and 5% water results in a reduction of approximately 30% in the savings of a pure graphite reflector. (auth)

1091 NAA-SR-Memo-719
North American Aviation, Inc., Downey, Calif.
SUGGESTED METHOD OF CONSTRUCTING THE THORIUM BOX ASSEMBLIES FOR THE U.P.R. C. J. Thompson. June 11, 1953. Decl. Mar. 27, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

1092 NAA-SR-Memo-1306
North American Aviation, Inc., Downey, Calif.
TRANSIENT PRESSURE IN THE WATER BOILER KINETIC EXPERIMENT. D. L. Hetrick and M. E. Remley. Mar. 18, 1955. Decl. Feb. 27, 1957. 23p. Contract AT-11-1-GEN-8. $0.30(OTS).

The results of theoretical studies on the effect of inertial pressures in water boiler transients are summarized as follows: (1) derivation of the equation for transient pressures in a partially enclosed container of water, (2) development of the equation of state for a mixture of water and radiolytic decomposition gas, (3) calculation of peak power as a function of reactor period, and (4) discussion of the effect of limited void space above the solution. It is intended that these results should serve as an indication of the magnitude of the inertial effects as a guide for planning the experimental program for KEWB. It is concluded that the attainable pressures represent no hazard to equipment, personnel or environs. (auth)

1093 NPG-16(Del.)
Nuclear Power Group, Chicago.
PRELIMINARY DESIGN AND COST ESTIMATE OF A HEAVY WATER MODERATED AND COOLED FIFTY MEGAWATT NUCLEAR POWER PLANT. SECTION 1. SUMMARY AND CONCLUSIONS. SECTION 2. DETAILS AND ANALYSIS. Nuclear Power Group, Chicago [Composed of American Gas and Electric Service Corp., Bechtel Corp., Commonwealth Edison Co.; Pacific Gas and Electric Co., and Union Electric Co. of Mo.]. Apr. 1, 1954. Decl. with deletions Apr. 15, 1957. 126p. $19.80(ph OTS); $6.30(mf OTS).

1094 ORNL-89
Oak Ridge National Lab., Tenn.
THE DETERMINATION OF PRESSURE DROP FACTORS THROUGH TYPICAL FUEL ELEMENT CHANNELS FOR HIGH TEMPERATURE GAS-COOLED THERMAL PILES. Charles L. Segaser. July 9, 1948. Decl. Mar. 4, 1957. 59p. $0.50(OTS).

The initial phase of the pressure drop testing program on proposed fuel element design shapes for the high-temperature, gas-cooled power reactor, has been completed with the compilation of pressure drop data for the flat-plate riveted-assembly type of unit, the ribbed cylinder type, the single bent-plate type both with and without rivets, the 13 hexagon-6 pentagon type, and the 7-hole graphite type. This phase of the program was conducted for the purpose of determining the equivalent skin-friction pressure drop factor for composite fuel element cross-sections, and for determining the joint loss factors for varying degrees of misalignment of the fuel units in the fuel channels of the pile. (auth)

1095 ORNL-794
Oak Ridge National Lab., Tenn.
MIXING IN THE CORE OF THE HRE. J. S. Culver, C. B. Graham, and R. H. Wilson. Nov. 13, 1950. Decl. Mar. 2, 1957. 24p. Contract W-7405-eng-26. $0.25(OTS).

Qualitative experiments indicate that the spiral flow in the outer portion of the core and the mechanical mixing in the center of the core are adequate to prevent boiling in the Homogeneous Reactor Experiment (HRE). Calculations, which assume no mechanical mixing in the center, show that evaporation below the boiling point in the central region will prevent boiling if provision is made to handle the large volumes of saturated gas evolved. An alternate design which provides for the injection of cold soup into the bottom of the vortex provides the best insurance against overheating the central region of the core and reduces the volume of gas to be removed. (auth)

1096 ORNL-1176
Nuclear Development Associates, Inc., White Plains, N. Y.
CRITICALITY CALCULATIONS FOR HYDROGEN MODERATED REACTORS FROM MICROSCOPIC DATA. Mathew Shapiro and Stanley Preiser. Nov. 29, 1951. Decl. Feb. 7, 1957. 26p. Contract AT(40-1)-1065 and Subcontract 394 under W-7405-Eng-26, for Oak Ridge National Lab. $0.30 (OTS).

The probability for a fission neutron to escape fast leakage from bare, hydrogen moderated reactors is calculated by a method utilizing only the microscopic cross section of the materials. These results can be employed to determine the effectiveness of other substances in preventing fast neutron leakage. These calculations are carried out for aluminum and iron. (auth)

1097 ORNL-1605
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JULY 31, 1953. W. E. Thompson, comp. Oct. 20, 1953. Decl. Mar. 2, 1957. 186p. Contract W-7405-Eng-26. $27.30(ph OTS); $8.10(mf OTS).

Progress on the Homogeneous Reactor Project is reported as follows: Part 1. Homogeneous Reactor Experiment. Details of experimental operation of the HRE at high power levels; concentration effect in the core due to gas removal; corrosion estimates from studies of stainless

steel test specimens; fuel recovery by flushing piping and storage system with 0.1M H_2SO_4 and 5% HNO_3; pump failure and replacement; estimated effect of stopping the fuel-circulating pump during operation; estimated response of the HRE to reactivity increases at low initial power; and calculated core pressures for power excursions. Part II. Boiling Reactor Research. Details of experiments to be performed on the Supo Reactor at Los Alamos to investigate kinetic behavior; further studies of nucleating characteristics of fission fragments in superheated and H-supersaturated solutions of UO_2SO_4; power-removal studies in 3-atm. pressure; effect of insertion of a large draft tube in a 6-ft-diam. H_2O system through which air was bubbled; and possible core arrangements for boiling homogeneous reactors. Part III. General Homogeneous Reactor Studies. Studies on the flowsheet, equipment, layouts, reactor vessel, heat exchanger, gas separator, catalytic recombiner, and condenser for the Intermediate Scale Homogeneous Reactor; nuclear calculations for one- and two-region U^{233} breeder reactors; summary of pump development and testing; control and instrumentation summary; summary of corrosion of various stainless steels, Ti, Co, Zr, and Ni alloys by oxygenated UO_2SO_4 and H_2O; results of two UO_2F_2-stainless steel loop runs; descaling of a stainless steel loop in which UO_2SO_4 and UO_2F_2 had been circulated; summary of program, methods, and equipment for future in-pile-loop corrosion tests; corrosion behavior of Malcomized and Stellite 21 coatings on stainless steel in oxygenated UO_2SO_4 solutions; corrosion behavior of ceramic materials in air-aerated UO_2SO_4 solutions; welding of austenitic stainless steels and corrosion of welds by UO_2SO_4; corrosion prevention by selective oxidation of stainless steel; impact and fatigue tests on commercial and annealed Ti specimens; phase studies on a Zr-In alloy; Al-Zr alloy phase studies; results of impact tests on irradiated Fe, carbon steel, stainless steel, and Ti; thermal oxidation of SO_2 by UO_2SO_4; exploratory studies on various Th blankets in aqueous solution; stability of and corrosion by irradiated UO_2SO_4 and UO_2F_2; neutron capture cross section of Pa^{233}; effects of U^{234}; U^{235}, and U^{236} buildup in core of U^{233} breeder reactors; preparation, thermal conductivity, and abrasive properties of Th oxide slurries; preparation of other Th slurry systems; properties of Th-U slurry systems; irradiation effects on Th oxide pellets; preparation and thermal conductivity of Th oxide; stability of Th hydroxide peptized with either Th nitrate or uranyl nitrate; preparation of homogeneous Th-U compounds; circulating loop for Th oxide slurries; circulating loop for UO_3 slurries; effect of Fe^{+3} and Ni^{++} additives on UO_3 slurries; circulation of other U slurry systems; solubility of CO_2 in H_2O as function of temperature and pressure; solubility of $CaSO_4$ in UO_2SO_4; results of batch-adsorption tests with CaF_2; and cost estimates for chemical processing of U^{233} breeder reactor blankets and cores. (L.M.T.)

1098 ORNL-1613(Rev.(Del.))
Oak Ridge National Lab., Tenn.
A CONCEPTUAL DESIGN OF A PRESSURIZED-WATER PACKAGE POWER REACTOR. A. L. Boch, W. R. Gall, G. F. Leichsenring, and R. S. Livingston. June 6, 1955. Decl. with deletions Jan. 24, 1957. 386p. Contract W-7405-eng-26. $1.75(OTS).

This conceptual design study describes a reactor and associated power plant that has been designed to produce 1000 kw of net electric power and 3535 kw of steam for heating purposes. The total thermal output of the reactor is 10,000 kw. The fuel plates consist of highly enriched UO_2 imbedded in a matrix of stainless steel and clad on all sides with stainless steel. The core is cooled and moderated with circulating light water, pressurized to 1200 psia. Saturated steam, produced in the heat exchanger at 200 psia, is used to drive a turbogenerator. Steam from the heat exchanger is also used, at a reduced pressure, for space heating. The reactor is loaded with approximately 18 kg of U^{235} and will supply 15 Mw-yr of energy before refueling is required. This corresponds to 2.5 years of operation at an average load factor of 60%. Burnout poison in the form of B_4C is incorporated to reduce the reactivity excursion and thus facilitate control. The major objective has been to design a reactor which will require a minimum of development effort and yet be reliable and inexpensive. The estimated capital investment, exclusive of uranium, is $1,703,000. The estimated cost per kilowatt-hour for net electric and steam power at the bus, based on a 60% average load demand is 5.33 cents and 1.23 cents, respectively. (auth)

1099 ORNL-1642(Del.)
Oak Ridge National Lab., Tenn.
AQUEOUS HOMOGENEOUS REACTORS FOR PRODUCING CENTRAL-STATION POWER. R. B. Briggs. May 21, 1954. Decl. with deletions Feb. 25, 1957. 105p. Contract W-7405-eng-26. $0.50(OTS).

Aqueous homogeneous reactors are examined from the standpoint of producing central-station power. Included are the materials requirements for a large power industry, a review of raw materials supplies, a status report on the technology of aqueous homogeneous systems, and a comparison of one- and two-region reactors that use Th and U fuels. (auth)

1100 ORNL-1753(Del.)
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1954. H. F. McDuffie, comp. Sept. 17, 1954. Decl. with deletions Apr. 17, 1957. 184p. Contract W-7405-eng-26. $27.30(ph OTS); $8.10(mf OTS).

Homogeneous Reactor Experiment. Final data were obtained on the effectiveness of Cu^{++} as an internal recombiner catalyst for radiolytic gas. Homogeneous Reactor Test. Criticality calculations have been completed for three blanket configurations using D_2O, ThO_2 slurry, and unenriched UO_2SO_4 solutions. Investigations on the temperature coefficient of reactivity and kinetic studies with respect to core pressure rise were also undertaken. Startup and shutdown procedures will involve the addition and removal of fuel concentrate. Revised flow sheet and design data sheets are presented, including the reactor vessel assembly, fuel pressurizer, recombiner-condenser, outer dump tank condenser, and fuel solution evaporator. The physical properties of HRT fuel and blanket solution at 2000 psia are given. Present evidence indicates that the Xe poison level can be maintained below 1% by continuous stripping with only 2% of the decomposition gases which would be produced if no Cu^{++} catalyst were present for recombination. Revised inlet and outlet valve designs for the HRT pulsafeeder system have resulted in improved performance. General Homogeneous Reactor Studies. Principal activities in engineering development consist of the design of several representative heat exchanger layouts and recombiner loops. An extensive body of data on the corrosion of stainless steel by UO_2SO_4 solutions has been accumulated. The corrosive effects of boiling HNO_3 on stainless steel, and the UO_2SO_4 corrosion of Zircaloy were also investigated. Stress relief annealing at 1000°F improved the dynamic corrosion resistance of austenitic stainless steel welds corroded by UO_2SO_4. The dynamic corrosion of Zircaloy-2 and Ti by UO_2SO_4 showed no marked effect on the impact behavior of these materials. The yield of N_2 from irradiated $Th(NO_3)_4$ is considerably less than previous

values reported for $UO_2(NO_3)_2$ solutions. A revised phase diagram of the UO_2F_2-H_2O system is presented. The preparation and corrosive properties of ThO_2 slurries were investigated, together with the abrasion characteristics on stainless steel and Ti. Considerations associated with chemical processing of fuel and blanket solutions, such as the solubility of rare-earth sulfates, are discussed. (T.R.H.)

1101 ORNL-1794(Vol.2(Del.))
Oak Ridge National Lab., Tenn.
THE OAK RIDGE NATIONAL LABORATORY RESEARCH REACTOR SAFEGUARD REPORT. VOLUME 2—APPENDIX. F. T. Binford, ed. Oct. 7, 1954. Decl. with deletions Mar. 18, 1957. 102p. Contract W-7405-eng-26. $15.30(ph OTS); $5.40(mf OTS).

A recommended uniform basis for designing the ORR shield is set forth. This includes design values for power level and emergent radiation, standard values for various material properties, and basic radiation intensities. (auth)

1102 ORNL-1834(Del.)
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR TEST. SUMMARY REPORT FOR THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS. S. E. Beall and S[idney] Visner. Jan. 5, 1955. Decl. with deletions. May 1957. 191p. Contract W-7405-eng-26. $28.80(ph OTS); $8.40(mf OTS).

The HRT has been examined with respect to possibilities for the escape of radioactive material from the reactor and its gas-tight enclosure. A complete description of the HRT is presented including flow diagrams, design, corrosion problems, components, shielding, plant layout, operating plans, and control. Every possibility for the leakage of active solution from the reactor piping and components has been examined. In addition to equipment failure from mechanical and corrosion causes, reactivity accidents and explosions have been thoroughly investigated. The confinement and detection of leaks are discussed. The hazards to the surrounding area in the event of release from the building of a large fraction of the activity in the reactor has been determined. The results of the study indicate that the likelihood of activity escape from the reactor is so small that the hazards to operating personnel and to the public compare favorably with normally accepted risks. (B.J.H.)

1103 SEP-210(Del.)
Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y.
FABRICATION OF HEATER TUBE FUSES. H. S. Kalish and S. I. Megeff. Apr. 12, 1956. Decl. with deletions Mar. 1, 1957. 17p. Contract AT-30-1-GEN-366. $3.30 (ph OTS); $2.40(mf OTS).

Various methods were investigated to produce a heater tube fuse. Utilizing powder metallurgy techniques, fuel tubes with an I.D. of 0.045 inch, O.D. of 0.075 inch, and length of $^{11}/_{16}$ inch, containing various compositions of copper + UO_2, were fabricated. Fabrication methods consisted of compacting the tubes in quarter-length units, sintering under optimum conditions, coining on a tin-plated inner tube with tin washers between units, and then heat treating to obtain a proper bond. By these techniques, 33 units containing various compositions of copper + $U^{235}O_2$ were fabricated and shipped to Atomics International for testing. In addition, work was performed on the brazing facet of the trigger element assembly, joining the copper + UO_2 fuel tube to a type 347 stainless steel thermal resistance pad and block. (auth)

1104 TID-10093
Tennessee Valley Authority. Div. of Power Supply, Chattanooga.
TVA NUCLEAR POWER COST STUDY—1954. Jan. 1955. Decl. Mar. 11, 1957. 100p. $0.60(OTS).

A survey was made of the basic characteristics of different reactor types and an estimation is presented of the power costs which might be expected in the future from various types of reactor plants, assuming an established nuclear power economy. The basic elements of nuclear power costs, and factors affecting fuel burnup cost are reviewed. Earlier conclusions are reaffirmed which postulate that an economical TVA power reactor will eventually become possible, that this point may be reached within ten years, and that TVA should continue to expand participation in the nuclear power program. (C.H.)

1105 WAPD-AD(M)-14
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
PARALLEL-FLOW, ROD-TYPE FUEL ELEMENT. DESIGN STUDY. E. Frisch. June 7, 1954. Decl. Mar. 18, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

The preliminary design study of a fuel assembly for the Shippingport reactor is presented. The design consists of a number of closely spaced, round, U—Mo rods, clad with Zircaloy and joined at each end to a rigid perforated plate. Several units would be stacked end to end to form a subassembly. (D.E.B.)

1106 WAPD-C-136
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
INVESTIGATION OF CLOGGING OF MARK I FILTERS; PRELIMINARY REPORT. Paul Cohen and V. W. Thompson. Apr. 20, 1953. Decl. Mar. 18, 1957. 20p. $3.30 (ph OTS); $2.40(mf OTS).

1107 WAPD-CP-1239
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
WATERLOGGED FUEL ELEMENT STUDY. R. M. Lieberman. 1955. Decl. Mar. 25, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

The purpose of this study is to determine the probability of rupturing a fuel element containing a given quantity of water when the element is subjected to heating. The results of a general theoretical study were compared with data obtained from a waterlogged sample broken in an autoclave. The findings were used to determine the probability of breaking of breaking the waterlogged fuel element which it is planned to insert into the HT-1 hole at the MTR. (auth)

1108 WAPD-CPM-M-19
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
PWR SIMULATOR STUDIES. SERIES IX. E. A. Wieczorkowski. Nov. 28, 1956. 50p. $7.80(ph OTS); $3.30(mf OTS).

Results of similator studies on (1) control system malfunctioning, (2) steam dump malfunctioning, (3) cold water accidents (full flow), and (4) normal variations are presented, with no attempt to analyze the results. (L.M.T.)

1109 WAPD-CPM-M-24
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
PWR SIMULATOR STUDY. SERIES VII. N. F. Simcic and E. A. Wieczorkowski. Dec. 5, 1956. 113p. $18.30(ph OTS); $6.00(mf OTS).

Differential equations, plant parameters, and design considerations used in simulating the PWR primary plant are presented. Results from various studies are presented as reproductions of recorder charts with no attempt made to analyze them. (L.M.T.)

1110 WAPD-FE-704
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
PROPOSAL FOR WAPD-NRX-CR-5 TEST OF U—Mo AND

UO_2 BUNDLES. K. M. Goldman. Mar. 17, 1955. Decl. Mar. 20, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

1111 WAPD-MDM-18

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

RESULTS OF SURVEY ON PWR FUEL ELEMENT END CLOSURE TECHNIQUES. Earl L. Richards. Aug. 15, 1954. Decl. Mar. 18, 1957. 22p. $4.80(ph OTS); $2.70 (mf OTS).

1112 WAPD-PWR-FEP-1146

[Westinghouse Electric Corp. Bettis Plant, Pittsburgh.] PRELIMINARY PROCESS SPECIFICATION FOR THE MANUFACTURE AND INSPECTION OF PWR FUEL ROD BUNDLES. T. J. Kisiel. Jan. 19, 1956. Decl. Mar. 18, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Specifications covering the manufacture and inspection of PWR fuel rod bundles are outlined. (L.M.T.)

1113 WAPD-RD-58

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PWR CORE (FUEL ROD SIZE). P. A. Halpine. June 17, 1955. Decl. Apr. 12, 1957. 6p. Contract AT-11-1-GEN-14. $1.80(ph OTS); $1.80(mf OTS).

Recommendations are made that the nominal fuel rod size be 0.413 in. in diameter and over-all rod length, including end caps be $10^1/_4$ in. A preliminary clad thickness of 0.028 in. was selected but is being further investigated. (L.M.T.)

1114 WAPD-ReL(W)-50(Del.)

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

INVESTIGATION OF A SEEDED PWR. Jan. 20, 1955. Decl. with deletions Mar. 11, 1957. 85p. $0.50(OTS).

An interim report is presented from a study of the thermal, hydraulic, and mechanical design aspects of seeded reactors as applied to pressurized water reactors. Annular seed arrangements are presently considered to be the most promising and are discussed in some detail. (C.H.)

1115 WAPD-ReM-4

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

FLOW DISTRIBUTIONS IN THE STR REACTOR VESSEL AS DETERMINED IN THE ARMOUR AIR TESTS. N. A. Petrick. Jan. 5, 1953. Decl. Feb. 25, 1957. 56p. $9.30(ph OTS); $3.60(mf OTS).

Tests were undertaken to determine the fluid flow characteristics of the Submarine Thermal Reactor reactor vessel. The tests were made using a half-size mahogany and aluminum model of the pressure vessel using atmospheric air as the working fluid. Schematic sketches and photographs of the test model are included, and test results are summarized. (C.H.)

1116 WCAP-120

Westinghouse Electric Corp. Commercial Atomic Power, Pittsburgh.

INTERNAL RECOMBINATION OF HYDROGEN AND OXYGEN. A Literature Review. J. R. McCord. Mar. 20, 1956. Decl. Mar. 15, 1957. 42p. $7.80(ph OTS); $3.30(mf OTS).

The purpose of this report is to summarize briefly the available information on the internal recombination of gaseous hydrogen and oxygen which will likely be either of value during experimental study or applicable to the design of the PAR. All the available quantitative data are included for the studies of recombination in slurries while only illustrative data are included for recombination in solutions. The bibliography was intended to be comprehensive (but is undoubtedly incomplete at present) and should be of value to anyone who is studying internal catalytic recombination. (auth)

Nuclear Transformation

1117 IDO-16002

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

MATERIALS TESTING REACTOR IRRADIATION OF THORIUM. C. F. Leyse. Nov. 9, 1951. Decl. Mar. 28, 1957. 72p. Contract AT(10-1)-205. $12.30(ph OTS); $4.50(mf OTS).

Various methods of irradiating Th in the MTR for production of U^{233} have been considered. The effects on reactor operation and experimental facilities have also been investigated. Production would come chiefly from Th placed in the reactor core shim-rod (shim-safety rod) positions and/or the Be reflector. The graphite reflector is not attractive for Th irradiation. The effects on experimental facilities are not prohibitive. (auth)

Particle Accelerators

1118 CRD-T2C-87

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

CONTAMINATION OF NaK COOLANT IN BARE THORIUM TARGET. L. M. Litz and R. E. Batzel. Mar. 13, 1952. Decl. Apr. 2, 1957. 18p. (LWS-24338). $3.30(ph OTS); $2.40(mf OTS).

The nature, magnitude, and sources of activities to be expected in the NaK coolant of the Th primary target for the Materials Testing Accelerator were estimated. The primary target consisted of bare Th plates inserted in a stainless steel tube bank with molten NaK as coolant. The major source of activity will be due to nuclear reactions induced in the NaK by the incident deuteron beam and slow neutron flux; fission-product recoil activities will be a secondary source of activity. (C.W.H.)

1119 LRL-71

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

MARK I MTA LINEAR ACCELERATOR RUN NO. 2. R. J. Vetterlein and P. V. Livdahl. Nov. 1953. Decl. Apr. 5, 1957. 15p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40 (mf OTS).

During Run No. 2, the first proton beam was accelerated A peak end-to-end cavity voltage of 25.1 million volts was attained with pulsed operation at a 16% duty cycle. Severe sparking at the 0-1 gap damaged the drift tube surfaces so that sufficiently high voltage could not be held for continued beam operation. The machine was shut down to clean drift tube surfaces again. Also the conical shaped, water-cooled Al target designed to absorb the full beam power was to be installed. (auth)

1120 LRL-75

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

MARK I MTA LINEAR ACCELERATOR RUN NO. 3. D. H. Birdsall, S. P. Stone, and A. K. Johnson. Dec. 1953. Decl. Apr. 5, 1957. 33p. Contract AT(11-1)-74. $6.30(ph OTS); $3.00(mf OTS).

Experimental operation of Mark I from July 29 through Sept. 24, 1952 is reported. During the preceding two months shutdown, spark-damaged drift tubes No. 0, 1, and

2 were repaired and chemically cleaned. After a slow "bake-in" procedure, a cavity voltage of 20-Mv at 16% duty cycle was obtained. Although operation was not too steady, some preliminary beam evaluation tests were made. Proton acceleration threshold voltage was determined to be 18.5 Mv, based on the associated rapid increase in neutron flux beyond the D-8B target. The beam was found to be about 1 in. off from the cavity axis at the target. A pulsed beam current of 50 to 70 ma peak was reached. The results indicated that there was nothing to prevent a high-current pulsed beam from being produced. However, in spite of careful cleaning and slow bake-in methods, sparking in the 0-1 gap was still a problem. Since small scale tests indicated that Inconel had better resistance to spark damage than the deoxidized high-phosphorous Cu originally used, Mark I was shut down to install Inconel overlays on the first two drift tubes. (For preceding report in series see LRL-71.) (auth)

1121 LRL-77

California Research and Development Co., Livermore Research Lab., Livermore, Calif.

MARK I MTA LINEAR ACCELERATOR RUN NO. 4. R. J. Vetterlein and P. V. Livdahl. Dec. 1953. Decl. Apr. 5, 1957. 51p. Contract AT(11-1)-74. $7.80(ph OTS); $3.30(mf OTS).

Mark I Linear Accelerator Run No. 4 covers the operation from Oct. 7 to Nov. 3, 1952, after a two weeks shutdown for the installation of Inconel overlays on the adjacent surfaces on drift tubes No. 0 and 1. Sparking was not eliminated by the use of Inconel, but immediate, full recovery of cavity voltage after a spark-down was almost always the case, thus permitting far more stable operation. The Inconel baked out rapidly and also resisted spark damage better than the DHP Cu. A 175-ma peak (over twice that obtained in Run No. 3) proton beam was accelerated with r-f voltage of 19 mv and a duty cycle of 1 to 2%. This low duty cycle was due to the inadequately cooled Inconel which was overlaid on the original water-cooled DHP surface. Beam operation was almost continuous for four days with very satisfactory operation. Several tests were made to determine the effect of various parameters, e.g., precessor voltage, injector pulse length etc., on the beam. The results of this run were very encouraging as steady operation with high beam currents, approaching the design expectation of 250 ma, was achieved. As sparking was still occurring, it was decided to shut down the accelerator and remove No. 1 drift tube to eliminate the shortest gap in the cavity. Operation under this condition was performed in Run No. 5. (auth)

1122 LRL-82(Del.)

California Research and Development Co., Livermore Research Lab., Livermore, Calif.

MTA PROJECT. TARGET AND PROCESS SECTION QUARTERLY PROGRESS REPORT [FOR] APRIL THROUGH JUNE 1953. Mar. 1954. Decl. with deletions Mar. 19, 1957. 72p. Contract AT(11-1)-74. $12.30(ph OTS); $4.50(mf OTS).

Nuclear chemistry studies of production characteristics of MTA-type targets were extended to NaK-cooled configurations. In the cases of target mockups composed of (a) Be primary–U secondary and (b) U primary–U secondary, it was found that the neutron yields internal to the targets were reduced by 30% as a result of the presence of the NaK that would be used for cooling. An analysis of high-energy nuclear events was completed for a solid uranium target bombarded by deuterons. The neutron yield with this target is about 33 neutrons per 500-Mev deuteron. Preliminary results with deuteron bombardment of Be, C, and NaK-primary targets–U secondary targets predict the smaller yields of 26, 23, and 22 neutrons per 500-Mev deuteron, respectively. Further studies were made on the feasibility of an MTA target using fuel in the form of a slurry suspended in liquid metal. Samples of various grades and treatments of Al alloys have been inserted in the MTR for an irradiation of approximately six months to determine the effect of neutron bombardment of potential cladding materials for target elements. Further work was done on the corrosion of Al and Zr by water. Corrosion tests of Al alloys coupled to stainless steel in water at 350°F showed essentially no difference in corrosion rates obtained with the Al alone. When coupled with Zr, a slight reduction in corrosion was observed. Studies continued on the chemical problems resulting from the use of the Purex process for separation of irradiated Zr-clad U target elements. A miniature mixer-settler was built, tested, and used successfully to simulate chemical behavior of Purex separations processes adapted to unique MTA fuel features. (auth)

1123 LRL-100

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

GRID DRIVEN POWER AMPLIFIER FOR MARK I MTA ACCELERATOR. O. A. Fredriksson, S. P. Stone, and V. B. Waithman. Apr. 1954. Decl. Apr. 10, 1957. 17p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

A driven power amplifier r-f system was installed on the Mark I linear accelerator for Run No. 10. The system, which used RCA A-2332 superpower, shield-grid triodes as final power amplifiers, is described. Problems associated with the initial excitation of the cavity and the techniques used to solve them are discussed. (auth)

1124 LRL-124

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

CHOICE OF RF VOLTAGE GRADIENTS FOR THE A-54 LINEAR ACCELERATOR. P. V. Livdahl, O. E. Myers, and R. J. Vetterlein. Apr. 1954. Decl. Apr. 10, 1957. 10p. Contract AT(11-1)-74. $1.80(ph OTS); $1.80(mf OTS).

Bases for the selections of gradients suitable for deuteron acceleration (0.5 Mv/ft and 0.55 Mv/ft) for the first two 1 $\beta\lambda$ cavities of A-54 linear accelerator are presented. Comparison is made of these gradients to pertinent experimental data and to voltage-holding phenomena experienced in Mark I and the Berkeley linear accelerator. It is shown that these gradients appear to be quite conservative and that proton gradient should be achieved with little r-f "bakeout" and no damaging sparks. (auth)

1125 LWS-24498

California Research and Development Co., Livermore, Calif.

NEUTRON DIFFUSION STUDY OF BASE CASE TARGET. Karl Bernstein. Aug. 13, 1952. Decl. Sept. 20, 1957. 14p. (CRD-T4A-125). $3.30(ph OTS); $2.40(mf OTS).

A two-medium two-group neutron diffusion study of the modified base case primary and secondary Materials Testing Accelerator target has been completed, using the latest U and water dimensions provided by the engineering group. This study provides information about the magnitude and distribution of Pu production and heat load within the target as well as the number of neutrons escaping from the target. Briefly, the results indicate that per incident deuteron, 9.9 atoms of Pu are formed within the target, 5.8 neutrons escape from the target, and 3.0 thermal fissions occur. (auth)

1126 LWS-24546
California Research and Development Co., Livermore, Calif.
STATUS REPORT OF HELIUM-COOLED, BAYONET TUBE, BERYLLIUM PRIMARY TARGET DESIGN STUDY. J. L. Smith. Aug. 15, 1952. Decl. Feb. 27, 1957. 23p. (CRD-T2B-153). $4.80(ph OTS); $2.70(mf OTS).

This study evaluates the heat transfer and fluid flow performance of a helium-cooled, unclad beryllium plate, bayonet tube primary target for the Materials Testing Accelerator. The predominant characteristics of this arrangement are the high plate and tube temperatures, high pumping power, and large target depths that are inherent for gas cooling. Another characteristic of the arrangement, revealed by this study, is the sensitivity of the pumping power requirements to changes of plate spacing and heat load. Some permissible target geometries were determined by preliminary header design, deuteron range studies, and the analysis of heat transfer and fluid flow performance. The estimated operating characteristics of these geometries are given in this report. Production studies indicate the relative external neutron yields of these arrangements compared with the water-cooled, clad uranium plate, bayonet tube primary target (base case). The advantages and disadvantages of the helium-cooled beryllium target compared to the base case primary target are presented. Recommendations are made concerning future work considered necessary for a complete evaluation of the theoretical design. Based on this study it appears that the proposed design may be operable and offer some advantages over a uranium primary. (auth)

1127 MTA-47
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
INJECTION FOR A HIGH CURRENT LINEAR ACCELERATOR. W. H. Gust and J. R. Rempel. Apr. 21, 1953. Decl. Apr. 3, 1957. 32p. Contract AT(11-1)-74. $6.30 (ph OTS); $3.00(mf OTS).

The MTA project requires an ion injector capable of dependably producing proton or deuteron peak beam currents of about one amp. A magnetically collimated linear type MTA ion injector is described which has produced 2.4 amp of 80 kv pulsed positive ions. Empirical alterations of ion source geometry which have improved injector operation and output are discussed. A toroidally wound transformer, which can continuously monitor beam currents ranging from 2.5 to 1300 ma without interfering with the beam, and a sectionalized beam current collector which provides beam profile information are also described. (auth)

1128 ORNL-1884
Oak Ridge National Lab., Tenn.
ELECTRONUCLEAR RESEARCH DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 20, 1955. F. T. Howard, ed. June 24, 1955. Decl. Feb. 22, 1957. 37p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

The 86-in. cyclotron is being modified to provide for deflection of the proton beam. Radioisotope production and cyclotron operation before shut-down are summarized. With the use of the 63-in. cyclotron, the absolute values of the electron capture and loss cross sections for elastic scattering of N by N was measured at energies from 13 to 22 Mev. A double-focusing 90° magnet is being designed for use in identifying the reaction products from N-induced nuclear reactions. The 44-in. cyclotron is being revised to provide for the acceleration of protons to 1.5 and 5 Mev. The feasibility of converting the 44-in. cyclotron to a 48-in. heavy-particle cyclotron is being studied, and design specifications are given. The production of Pu isotopes by electromagnetic separation, Pu recycle chemistry, and product processing are discussed. The Army Package Power Reactor program is summarized. APPR-type fuel assemblies have been fabricated for irradiation experiments and are being corrosion tested. Feasibility studies of a fixed-frequency 1-bev accelerator are reported. (W.L.H.)

Radiation Absorption and Scattering

1129 CF-52-4-99
Oak Ridge National Lab., Tenn.
SOME GROUND SCATTERING EXPERIMENTS PERFORMED AT THE BULK SHIELDING FACILITY. H. E. Hungerford. Apr. 16, 1952. Decl. Feb. 22, 1957. 20p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Graphs are presented of the ground scattering of both neutrons and γ rays as a function of the distance above the ground. The ground used in the experiments was either water or concrete. Cross sectional diagrams are presented of the arrangement of the apparatus for both the neutron and γ-ray experiments. (M.H.R.)

Radiation Effects

1130 ANL-5249
Argonne National Lab., Lemont, Ill.
DIFFERENTIAL THERMAL ANALYSIS OF IRRADIATED DIAMOND AND SILICON CARBIDE. William Primak. Dec. 1, 1954. Decl. Feb. 14, 1957. 22p. Contract W-31-109-eng-38. $4.80(ph OTS); $2.70(mf OTS).

It was demonstrated by differential thermal analysis (DTA) that: 1. Catastrophic amounts of energy can be stored in diamond. 2. Even at low irradiations, the release takes place over several hundred degrees, indicating a spectrum of activation energies. 3. At higher irradiations, the stored energy release is considerably less than the increased energy content and seems not to have been completely released even at the highest temperatures reached. 4. There is some indication of an increased heat capacity below the temperature of stored energy release. It was shown by DTA that large amounts of energy can be stored in silicon carbide on irradiation. The release was found to be spread out over a greater range of temperatures than in diamond and indicated a larger and higher group of activation energies. Catastrophic release was not achieved. The amount of stored energy released over the range of temperatures used was 140 cal/g in a sample irradiated in a water-cooled testhole at HEW for an exposure of 265 Mwd/aT. (auth)

1131 CF-50-7-5
[Oak Ridge National Lab., Tenn.]
THEORY OF RADIATION DAMAGE. D. K. Holmes. July 6, 1950. Decl. Apr. 2, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An investigation was made of the causes of macroscopic property changes in materials exposed to pile radiation. The contribution made by collisions between ions, and between ions and neutrons are considered in order to obtain general principles from which an estimate may be made of the damage to be expected in any given situation. Tabulated data are included on graphite, aluminum, and uranium metal. (C.H.)

1132 HW-27862
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

THERMAL SPIKES AND URANIUM INSTABILITY. G. A. Last and D. McLachlan, Jr. Dec. 10, 1952. Decl. Feb. 19, 1957. 13p. Contract W-31-109-Eng-52. $0.25(OTS).

A possible mechanism for the instability of U in radiation is sought in the deformations accompanying a thermal spike in solid metal. On the basis of this mechanism, the behavior of U under irradiation is predicted. (T.R.H.)

1133 KAPL-M-AHW-4
Knolls Atomic Power Lab., Schenectady, N. Y.
FACTORS AFFECTING THE IRRADIATION BEHAVIOR OF FISSIONABLE MATERIALS UO_2, UZr, AND B. A. H. Willis, G. F. McKittrick, and E. P. Renzi. May 2, 1956. Decl. Mar. 12, 1957. 14p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Data on the volume increases upon irradiation of U–Zr, B^{10}–Zircaloy-2 and B^{10}–Ti are given. Generally, the order of magnitude of volume change per % burnup of the total core atoms is predictable. (T.R.H.)

1134 KAPL-M-GDC-1
Knolls Atomic Power Lab., Schenectady, N. Y.
A PROGRAM FOR EVALUATING NUCLEAR RADIATION STABILITY OF ELECTRICAL INSULATION. E. L. Minoher and G. D. Collins. Apr. 20, 1951. Changed from OFFICIAL USE ONLY Jan. 28, 1957. 13p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

An investigation was made to determine suitable electrical insulations for use in liquid metal pumps which must provide at least 40,000 hours of service, and in the windings of three-phase, alternating current motors to be used aboard a naval vessel. The requirements of service are given. The properties required of insulations, and the methods to measure them are described. (auth)

1135 ORNL-1818
Oak Ridge National Lab., Tenn.
PRODUCTS PRODUCED IN BATCH NEUTRON IRRADIATION OF THORIUM. A. T. Gresky and E. D. Arnold. Dec. 27, 1955. Decl. Mar. 19, 1957. 39p. Contract W-7405-eng-26. $0.40(OTS).

Calculated data and graphs describing the effects of batch thermal-neutron irradiation of thorium, the usual method of operation of heterogeneous reactors, are presented. The buildup and decay of U^{233}, Pa^{233}, other heavy isotopes, and fission products are considered on the basis of best available cross-section and fission-yield data. The effects of these irradiation products on the Thorex chemical separation process are indicated briefly. (auth)

1136 ORNL-2039(Del.)
Oak Ridge National Lab., Tenn.
FISSION PRODUCTS ACTIVITIES IN IRRADIATED NATURAL URANIUM, ENRICHED URANIUM, AND THORIUM. E. D. Arnold. Mar. 28, 1956. Decl. with deletions Mar. 4, 1957. 31p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

Calculated data and graphs describing the effects of batch thermal-neutron irradiations on the buildup of fission products in natural uranium, enriched uranium, and thorium are presented together with empirical equations and plots correlating total fission product activities and/or decontamination factors. (auth)

1137 TID-10017
Battelle Memorial Inst., Columbus, Ohio.
EFFECT OF PARTICLE SIZE FISSION-FRAGMENT DAMAGE FOR PARTICLES OF FISSIONABLE MATERIAL DISPERSED IN A MATRIX. Ralph J. Harrison. June 30, 1953. Decl. Mar. 18, 1957. 42p. Contract W-7405-eng-92. $0.35(OTS).

Because of the limited range of fission fragments, a matrix containing fissionable material in the form of a particle dispersion will show less damage in a thermal-neutron flux than a matrix containing the same amount of effect is discussed quantitatively with application to resistivity, using a model involving simplifying assumptions of both geometrical and physical character. The theory is divided into four steps, each of which is discussed separately, with results given in the form of graphs and equations. Examples are discussed illustrating the application of the theory. (auth)

1138 WAPD-78(Del.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
EFFECTS OF NEUTRON BOMBARDMENT UPON THE PROPERTIES OF ZIRCONIUM AND OF A ZIRCONIUM–TIN ALLOY. M. L. Bleiberg and L. S. Castleman. Mar. 17, 1953. Decl. with deletions Mar. 4, 1957. 35p. Contract AT-11-1-gen-14. $6.30(ph OTS); $3.00(mf OTS).

Changes in properties of unalloyed crystal bar zirconium and of a 2.5% tin–crystal bar zirconium alloy were determined after irradiations at process water temperatures in the MTR. Increases in electrical resistivity and in hardness were observed, which appeared to reach saturation values after sufficiently long exposures. The notch toughness of irradiated specimens were apparently somewhat improved after short exposures, but unchanged after longer exposures. (auth)

Radioactivity

1139 LA-1958(Del.)
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF FISSIONABLE METALS AS BASIC NUCLEAR SAFETY DATA. H. C. Paxton and Glen A. Graves. Apr. 1956. Decl. Sept. 7, 1956. 18p. Contract W-7405-Eng-36. $0.30(OTS).

Data on critical configurations of fissionable metals are summarized in a form emphasizing the influence of conditions commonly of concern in nuclear safety questions. Although the bulk of the data is derived from experiments with uranium metal enriched, usually, to about 90 wt. % U^{235} there are enough data for Pu and U^{233} to establish some general relations about their criticality relative to U^{235}. The specific information includes the influences exerted upon critical mass by various reflectors, by shape of the fissionable material, by variations in U^{235} concentration, by variations in material density, and by graphite and hydrogenous diluents. (auth)

Rare Earths and Rare-earth Compounds

1140 KAPL-M-JRS-10
Knolls Atomic Power Lab., Schenectady, N. Y.
CROSS-SECTION ESTIMATES FOR DYSPROSIUM. J. R. Stehn. June 19, 1956. 13p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Dysprosium is one of a number of materials that could be used as a burnable poison, for at least one of its isotopes has a neutron absorption cross-section that is greater than the fission cross-section of U^{235}. The information available to date on this element is considerably less extensive than on most elements. This memo gives a summary of our present cross-section knowledge on Dy and an estimate of the uncertainties therein. It is impossible to conclude from what we now know whether or not Dy has the nuclear properties requisite to be a successful burnable poison. Two experiments to elucidate the situation are suggested. (auth)

Shielding

1141 ANL-4864
Argonne National Lab., Lemont, Ill.
CP-10 SHIELD DESIGN. M. Grotenhuis and J. W. Butler. June 1, 1952. Decl. Feb. 12, 1957. 37p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

The design calculations for a reactor shield consisting of 2 ft of graphite, a sheet of boral, 6 in. of Fe, and 7.25 ft of ordinary concrete are given. (T.R.H.)

1142 KAPL-M-RCG-2
Knolls Atomic Power Lab., Schenectady, N. Y.
GENERAL SHIELD THICKNESS EQUATIONS AND CURVES. R. C. Gerber, Jr. Feb. 1, 1950. Decl. Feb. 28, 1957. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The method of calculating shield thickness currently in use at KAPL is described, and some equations were plotted to facilitate future work. (F.S.)

1143 ORNL-503
Oak Ridge National Lab., Tenn.
TECHNICAL SUMMARY OF ALL ORNL SUMMER SHIELDING SESSION (1949). Ralph Zirkind. June 19, 1950. Decl. Feb. 22, 1957. 50p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

An evaluation was made of the theoretical and experimental aspects of both neutron and gamma reactor shielding. Basic data are reviewed and analytical and numerical methods are discussed. Neutron cross-section data and gamma-ray absorption coefficients of particular interest in shielding studies are included. (C.H.)

Uranium and Uranium Compounds

1144 CF-55-4-30(Del.)
Oak Ridge National Lab., Tenn.
GAMMA ACTIVITY OF U^{233} DUE TO U^{232} DAUGHTERS. R. E. Leuze. Apr. 4, 1955. Decl. with deletions Mar. 12, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

1145 K-740(Del.)
Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.
A TEST OF NEUTRON MULTIPLICATION BY SLIGHTLY ENRICHED URANIUM. Dixon Callihan, D. F. Cronin, R. L. Macklin, J. W. Morfitt, and D. V. P. Williams. Mar. 28, 1951. Decl. with deletions Mar. 13, 1957. 12p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

A test has been made of the nuclear safety of a cylindrical container, 8 ft. high and $2\frac{1}{2}$ ft. in diameter, filled with approximately 3600 kg of UF_6 having a U^{235} enrichment of 2% and a U^{235} content of about 50 kg. The vessel could be enclosed by at least $8\frac{1}{2}$ in. of water although essentially no hydrogen was mixed with the uranium. No significant neutron multiplication was observed indicating the system to be far from critical, however the results are not sufficiently quantitative to allow an estimation of the critical conditions by extrapolation. Two cylinders of 2% enriched UF_6, each $2\frac{1}{2}$ in. in diameter and 4 ft. high, with axes parallel and sides in contact were also not critical when immersed in water. (auth)

1146 KAPL-M-HBS-4(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
METHODS FOR WEIGHING URANIUM IN URANIUM–ALUMINUM ALLOY FUEL DISKS. Hugh B. Stewart. Dec. 29, 1953. Decl. with deletions Feb. 26, 1957. 11p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Measurements made by weighing Al–U disks in air and in oil provide some information on the density of U. Reactivity coefficient type measurements were made in the internal thermal column, but this is a rather slow method. A program was proposed for a method for weighing the U in Al–U alloy fuel disks. (F.S.)

NUMERICAL INDEX OF REPORTS

This is a list of reports abstracted in this issue of *Nuclear Science Abstracts*, with the corresponding abstract number reference.

The report number codes listed below apply to AEC reports unless otherwise designated. Other codes are listed as British, Canadian, or Non-AEC. Those codes marked Misc. are used for both AEC and non-AEC reports. For a complete description of all report number codes used by the AEC, see TID-85, *Report Number Series Used by the Technical Information Service in Cataloging Reports*. September 1957. 112p. $3.00 (OTS). Some reports appear under more than one code designation and wherever practical all report numbers are listed in this index.

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